DOCUMENT RESUME

ED 192 091 CE 026 649

TITLE	Nuclear Technology Series. Course 7: Reactor Operations.
INSTITUTION	Center for Occupational Research and Development, Inc., Waco, Tex.: Technical Education Research Center, Waco, Tex.
SPONS AGENCY	Office of Education (DHEW), Washington, D.C.
PUB DATE	Jun 80
CONTRACT	OEG-0-74-1679
NOTE	209p.: For related documents see CE 026 639-672, CE 026 779-780, CE 026 782-785, and CE 026 855-856.
AVAILABLE FROM	Center for Occupational Research and Development, 4800 Lakewood Dr., Suite 5, Waco, TX 76710 (Modules are available individually or by course: write for prices)
EDRS PRICE	MF01 Plus Postage. PC Not Available from EDRS.
DESCRIPTORS	*Administration: Learning Activities: Learning Modules: *Nuclear Energy: Postsecondary Education: Quality Control: *Technical Education: Technical Occupations
IDENTIFIERS	Nuclear Energy Occupations: *Nuclear Power Plants: *Nuclear Reactors: Nuclear Technology

ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Reactor Operations, contains seven modules: Reactor Operator Licenses, Administrative Control for Nuclear Systems, Reactor Operating Experiences, Reactor Startup Procedures, Pressurized Water Operating Characteristics, Boiling Water Reactor Operating Characteristics, and Refueling of Reactors. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 956 and CE 026 639-643.) (LRA)

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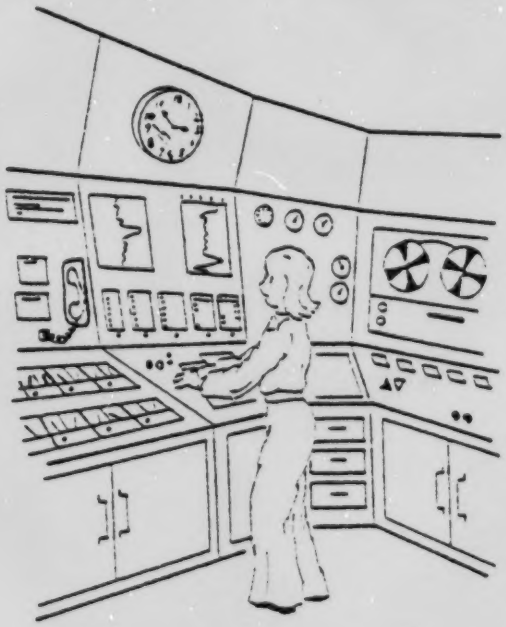
NUCLEAR TECHNOLOGY

Series

MODULE 7-1 REACTOR OPERATOR LICENSES. -

JAMES R. PENDLAND



REACTOR OPERATION

COURSE 7

U.S. DEPARTMENT OF HEALTH, EDUCATION & WELFARE
NATIONAL INSTITUTE OF EDUCATION

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TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER - SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 7 REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

The Nuclear Regulatory Commission (NRC) is charged with ensuring that only competent and dependable persons are granted licenses as operators of nuclear reactors (Atomic Energy Act of 1954, amended by the Energy Reorganization Act of 1974).

In 1976, NRC issued an "NRC Operator Licensing Guide," NUREG-0094, Rev 1 of WASH 1094, which describes the procedures and criteria for license issuance, which assists applicants to better understand provisions of the Law.

This module acquaints the student with the licensing process.

MODULE PREREQUISITES

None.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Cite number of the Federal regulation that gives NRC authority to regulate operator licensing.

2. List and explain areas of inquiry contained in the license application.
3. List elements of 10CFR55.11 that must be satisfied before a license is granted.
4. List areas covered in operator's written examination.
5. List additional areas covered in senior operator examination.
6. List areas covered in operating examination.
7. List basic limitations on an operator license.
8. State duration of a license, and condititions required for renewal.

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Module 7-1 Reactor Operator Licenses

Module 7-2 Administrative Control for Nuclear Systems

Module 7-3 Reactor Operating Experiences

Module 7-4 Reactor Startup Procedures

Module 7-5 Pressurized Water Reactor Operating Characteristics

Module 7-6 Boiling Water Reactor Operating Characteristics

Module 7-7 Refueling of Reactors

COURSE 7
REACTOR OPERATIONS

TECHNICAL EDUCATION RESEARCH CENTER - SW

4201 LAKE SHORE DRIVE - SUITE 111

WACO , TEXAS 76710

JUN 1980

CE 026 649

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PREFACE

ABOUT NUCLEAR TECHNOLOGY MODULES

The modules - developed by TERC-SW for use in two-year postsecondary technical institutions to prepare technicians for employment - are useful in industry for updating employees in company-sponsored training, and as laboratory texts in engineering schools. The principles, techniques and skills taught in the modules, based on tasks that nuclear technicians perform, were obtained from a nationwide advisory committee of employers of nuclear technicians. Each module was written by a working technical expert and tested with students.

A module contains the following elements:

Introduction, which identifies the topic and often includes a rationale for studying the material.

Module Prerequisites, which identify the material a student should be familiar with before studying the module.

Objectives, which clearly identify what the student is expected to know for satisfactory module completion. The objectives, stated in terms of action-oriented behaviors, include such action words as operate, measure, calculate, identify and define, rather than words with many interpretations, such as know, understand, learn and appreciate.

Subject Matter, which presents the background theory and techniques supportive to the objectives of the module. Subject matter is written with the technical student in mind.

Exercises, which provide practical problems to which the student can apply this new knowledge.

Materials, which identify the equipment required to complete the laboratory procedure.

Procedures, which is the experiment section, or "hands-on" portion of the module (including step-by-step instruction) designed to reinforce student learning.

Data Tables, which are included in most modules for the first year (or basic) courses to help the student learn how to collect and organize data.

Reference Materials, which are included as suggestions for supplementary reading/viewing for the student.

Test, which measures the student's achievement of the prestated objectives.

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The Nuclear Technology Series will consist of five curricula when completed:

Nuclear Reactor (Plant) Operator Trainee.
Radiation Protection Technician.
Nuclear Materials Processing Technician.
Nuclear Quality-Assurance/Quality-Control Technician.
Nuclear Instrumentation and Control Technician.

The completed Nuclear Reactor (Plant) Operator Trainee curriculum consists of the following sixteen technical specialty courses made up of 114 instructional modules:

Radiation Physics.
Radiation Protection I.
Principles of Process Instrumentation.
Heat Transfer and Fluid Flow.
Introduction to Nuclear Systems.
Instrumentation and Control of Reactors and Plant Systems.
Reactor Operations.
Reactor Safety.
Reactor Auxiliary Systems.
Power Plant Systems.
Radiation Detection and Measurement.
Reactor Physics.
Power Plant Chemistry.
Introduction to Quality Assurance/Quality Control.
Metallurgy and Metals Properties.
Mechanical Component Characteristics and Specifications.

Availability: Instructional modules are available individually or by courses from:

Technical Education Research Center - Southwest
4201 Lake Shore Dr., Suite 111
Waco, Texas 76710
Telephone (817) 772-8756

Printed:

AUTHORITY

Title 10 Code of Federal Regulations Part 50 Section 50.54(i) [10CFR50.54(i)] prohibits the manipulation of the controls of a nuclear facility by anyone not licensed as an operator or senior operator.

10CFR55.3 prohibits anyone not licensed from performing the duties of an operator or senior operator.

10CFR55.4(f) defines controls as "apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor."

EXCEPTIONS

There are, however, some exceptions to the requirement of a license. 10CFR55.9 exempts a student who manipulates the controls as part of a nuclear engineering course or as part of a training program to qualify for an operator license. In both of these situations, the individual must be under the direction and in the presence of a licensed operator or senior operator.

Also under 10CFR55.7, the NRC may grant other such exemptions from the licensing requirements that it feels are authorized by law, will not endanger life or property and are otherwise in the public interest.

In most cases, however, exemptions are not granted, and the person wishing to become an operator must file the

required application form, undergo a medical evaluation and demonstrate competence in the operation of the facility for which the license is requested.

APPLICATION

It is the responsibility of the individual and the facility at which the individual is employed to request NRC to grant a license. It is necessary that both the individual and the facility request the license; a license should not be requested if the individual doesn't want one, and the facility must intend to utilize the applicant's services.

CONTENT

The application form is divided into two broad areas of inquiry and attached to a medical evaluation form. The subject matter of the first part of the application is personal data of the applicant, and the second is the data pertaining to the facility for which the license is sought.

Personal Data

This information describes the individual requesting the license and includes biographical data, such as age, education and experience pertinent to granting an operator license.

Facility Data

This information includes name and location of facility, written request by a representative of the facility for an operating test, evidence that applicant has learned to operate reactor in a safe and competent manner and a statement that license is needed.

Medical Evaluation

Attached to the application is a medical evaluation of applicant. Form NRC-396, "Certificate of Medical Examination," which, completed by a licensed physician, outlines physical condition and general health of applicant. It is important that the administering physician completely fill in the application. Otherwise, the NRC will return it.

APPROVAL

10CFR55.11 describes the three-part test for approval of application as follows:

"The physical condition and general health of the applicant are not such as might cause operational errors endangering public health and safety." This requirement is designed to ensure that the reactor is not operated by an operator who is physically unable to perform the duties and that an individual subject to sudden incapacity is not put into a position where such incapacity would leave the reactor without an operator.

Applicant has passed a written examination and operating test to determine that the applicant has learned

to operate the reactor in a safe and competent manner. The purpose of such a two-part examination is to ensure competence on both theoretical and practical levels.

The applicant's services will be utilized on the facility for which the license is requested.

EXAMINATIONS

The scope of the written and operating tests for a license as an operator or senior operator includes an understanding of the facility design, familiarity with the controls and the operating procedures of the facility. While the operating test is generally similar for both operator and senior operator applicants, the written exam for the senior operator is broader in scope and deeper in detail.

OPERATOR'S WRITTEN EXAMINATION

The written exam, which the applicant is required to successfully complete for an operator's license, includes:

Reactor theory.
General design features.
Mechanical design features.
Auxiliary systems.
General operating characteristics.
Reactivity control mechanisms and instrumentation.
Safety systems.

Emergency systems.
Standard and emergency procedures.
Radiation monitoring systems.
Radiological safety principles.

While each of these areas is covered in a technologist curriculum (such as the one in which the student is presently engaged), the examination details a particular facility at which the license will be used. For this reason, the granting of a license to operate one plant does not grant an unlimited license to operate different plants.

SENIOR OPERATOR'S WRITTEN EXAMINATION

While it should be obvious that a senior operator must be able to successfully take an operator's exam, the senior operator's exam must go farther and into more detail. In addition to the subjects covered in the operator's exam, the following items appear in the senior operator's written exam:

Conditions and limitations placed on facility license.
Design and operating limitations in technical specifications, as issued by NRC.
Procedures for design and operating changes.
Radiation hazards that arise during unusual operating conditions.
More detailed reactor theory.
Specific operating characteristics, including effects on plant operation of temperature, pressure and reactivity changes.
Procedures and limitations, involving core geometry and control-rod programming.

Fuel handling.
Radioactive waste disposal.

OPERATING TEST

Since the reactor operator may be required to perform all of the approved operations associated with a nuclear reactor, the scope of the operating exam for both an operator and a senior operator are similar. In general, an operator must be able to operate a reactor and perform all associated functions. Thus, the applicant must be able to demonstrate a knowledge of the following:

Pre-startup procedures.
Manipulation of controls to establish and maintain reactor power level from shutdown to full power.
Instrumentation systems and indication associated with them (also, proper action for alarm conditions).
Behavioral characteristics associated with facility.
Design and operation of heat removal systems under normal and emergency conditions.
Operation of facility auxiliary systems, and their potential effect on reactivity.
Radiation monitoring systems, including fixed and portable instruments, and personal monitoring devices.
Facility's emergency plan, including duties and responsibilities of facility personnel.
Need for a careful and deliberate approach to responsibility associated with safe operation of facility.

During this exam, applicant may be required to perform duties of an operator while being observed by an NRC representative. Any subject which is germane to the facility'

operation is fair game and should be taken seriously. The examiner is guided by the criteria of ensuring that the operator is capable of handling both normal and abnormal conditions that arise from time to time.

WAIVER OF EXAMINATIONS

Contrary to the general rule that an applicant for a license must pass written and operating exams, the Commission may, upon request, waive part or all of this requirement. As expected, this waiver will generally be granted only in cases involving individuals with extensive operating experiences at a facility comparable to the one for which the waiver is requested.

10CFR55.24 also requires that the experience be within the last two years. As part of the request for such a waiver, the applicant is required to show a personal history of competence and safe operation at the former facility, the extent of those duties and that the applicant has learned to competently and safely operate the present facility.

EXAMINATION PRIOR TO CRITICALITY

A problem arises, when a new nuclear facility is being constructed, in regard to granting licenses for operators who will perform initial testing and operations. The Commission has made provisions for conducting an operating exam by simulation (as set out in section 55.25).

DENIAL OF LICENSE REQUEST

If a license request is denied for failure to satisfy one or more requirement, the applicant has certain steps available for reconsideration of the application.

In the case of a medical related denial, the applicant may submit additional related information for NRC consideration.

For failure of the written or operating examinations, applicant may file a new application two months from the date of denial; a third application may be filed six months after a second denial, and all other applications may be filed two years after the date of denial of each prior application.

If only the written or the operating examination is failed but the other is passed, the Commission may, upon request, excuse the applicant from that part which was passed.

ISSUANCE OF THE LICENSE

Once an applicant has satisfied the requirements set forth by NRC, a license will be issued to the applicant for a usual term of two years; however, the license is subject to any conditions the Commission determines are required to protect health and minimize danger to life or property. Limitations include:

License cannot be transferred to another individual.

License is limited to facility for which it is issued.

License is limited to specific controls specified in license.

License is subject to all NRC rules, regulations and orders.

Operator must demonstrate to NRC a present satisfactory knowledge and understanding of facility if functions of an operator have not been performed during a period of four months.

While the general duration of effectiveness of an operator's license is two years, the Commission may revoke, suspend or modify any license for the following reasons:

A materially false statement appears on the application.

Operator fails to comply with conditions, rules, regulations or orders of Commission.

Operator engages in conduct the Commission considers potentially hazardous to operation of the facility.

RENEWAL OF AN EXISTING LICENSE

Prior to the expiration of an existing license, each operator or senior operator must apply to NRC for license renewal for another two-year term.

As a general rule, the license will be renewed if the following five-part test has been met:

Applicant's condition and general health will not cause operational errors endangering public health or safety.

Applicant has been operating in a safe and competent manner.

Applicant is capable of continued safe and competent operation.

Applicant has completed or is engaged in an approved requalification program.

Applicant has continuing need for license.

If applicant makes a renewal application prior to thirty days before license expiration, license does not expire until NRC has made a determination on renewal request.

REQUALIFICATION PROGRAM

Appendix A of 10CFR55 requires that each operator demonstrate continued competence every two years, either by a reexamination or by participation in an approved requalification program, or both. The reason for such a requirement is to ensure the competence of the operators under all conditions, but especially under abnormal or emergency situations.

SUMMARY

Issuance of an operator license is not a simple matter. NRC has a serious obligation to ensure the same and competent operation of a growing number of nuclear power plants. As a result of this obligation, the Commission requires that the applicant demonstrate that future duties will be per-

formed in a highly competent manner in accordance with all rules, regulations and orders of the Commission.

EXERCISES

A copy of NUREG 0094, "NRC Operator Licensing Guide," will be provided by instructor. Review all seven sections of IV. "Content of Operator Written Examination;" become familiar with categorical groupings for the written examination.

MATERIALS

One copy of NUREG 0094, "NRC Operator Licensing Guide."

PROCEDURES

1. Listed in the Appendix is an extract from NRC publication, "NUREG 0094." Review each of the "Typical Sam-

ple Questions" and answer those given by instructor so that you are familiar with the type of questions appearing in the operator's written examination.

2. Listed in the Appendix is a copy of NRC form 157, "Examination Report." Familiarize yourself with the checklist used by NRC examiners during the examination.

DATA TABLE

None.

REFERENCE MATERIALS

American National Standards Institute (ANSI). "Selection and Training of Personnel for Nuclear Power Plants." ANSI Standard 18.7. New York: ASME, 1973.

U.S. Atomic Energy Commission. Title 10 - Atomic Energy, Code of Federal Regulations, Part 55, "Operator's Licenses."

U.S. Nuclear Regulatory Commission. "NRC Licensing Guide," No. NUREG 0094, Revision 1 of WASH 1094. July 1976.

AEC and NRC documents are available by writing the Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

APPENDIX

Contains Appendices D, E, F and G
extracted from NUREG-0094, revision 1
of WASH 1094 by permission of the
U.S. Nuclear Regulatory Commission

Form NRC-396
(3-79)
10 CFR 55

APPENDIX D

Form approved,
Bureau of Budget No. 38-R1004

U. S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF MEDICAL HISTORY

Facility Operator's or Senior Operator's License

Instructions: Applicant must complete all items on page 1. Typewrite or print in ink. Physician must complete all items on page 2.

1. Last Name | First Name | Middle Name | 2. Date of Birth

3. Home Address | 4. Sex

Have you ever had or do you now have any of the following? Give details of any condition answered in the affirmative under item 37.

	Yes	No		Yes	No
5. Rheumatic fever			16. Bone, joint, or other deformity		
6. Frequent or severe headaches			17. Painful or "trick" shoulder		
7. Dizziness or fainting spells			18. Painful or "trick" elbow		
8. Eye trouble			19. Paralysis		
9. Diabetes			20. Epilepsy or fits		
10. Tuberculosis			21. Depression or excessive worry		
11. Chronic shortness of breath			22. Loss of memory or amnesia		
12. Pain or pressure in chest or "heart attack"			23. Nervous condition which could impair judgment or reliability		
13. High blood pressure					
14. Low blood pressure			24. Drug, narcotic habit or excessive drinking		
15. Peptic ulcer			25. Do you normally wear eyeglasses?		

Complete each of the following. Give details of every affirmative answer under item 37.

	Yes	No
26. Has your work ever been limited or restricted for medical reasons?		
27. Have you ever been denied or rated up for life insurance for medical reasons?		
28. Have you ever been under observation or received care or treatment for any mental or nervous condition as a patient in a hospital, sanitorium, clinic, or other facility, or from a physician, clinical psychologist, etc.?		
29. Have you ever been rejected for or discharged from employment or military service for physical, mental, or nervous disorder reasons?		
30. Have you ever received, is there pending, have you applied for, or do you intend to apply for pension or compensation for existing disability?		
31. Have you ever seriously considered committing suicide?		
32. Have you ever been convicted of any violation of Federal law, State law, county or municipal law, regulations or ordinance? Do not include anything that happened before your 16th birthday. Do not include violations for which a fine of $25 or less was imposed.		
33. Have you ever had any major illness or injury other than those already noted?		

34. How many jobs have you had in the last 3 years?

35. What is the length of time in your present employment?

36. Give a brief statement of your present health in your own words:

37. Details of any items 5 through 33 answered in the affirmative. In addition, if your medical history includes any matter relating to physical, mental, or nervous condition, please describe the condition and set forth your explanation of why this matter would not affect your ability to function as a facility operator. Use additional sheet if more space is needed.

38. I certify that the foregoing information supplied by me is true to the best of my knowledge, and authorize the Nuclear Regulatory Commission to use any of the information in this certificate in the exercise of its authority over the licensing of operators.

(Date) | (Facility) | (Signature of applicant)

SIGN YOUR NAME IN INK AS IT APPEARS ON YOUR APPLICATION FOR OPERATOR'S OR SENIOR OPERATOR'S LICENSE

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MEDICAL EXAMINATION

39. Doctor: It is essential that each of the items on this page be completed. Sign the Certificate and mail to the Director, Division of Reactor Licensing, United States Nuclear Regulatory Commission, Washington, D.C. 20555 for reactor operator license applicants or to the Director, Division of Materials and Fuel Cycle Facility Licensing, United States Nuclear Regulatory Commission, Washington, D.C. 20555 for applicants for non-reactor licenses.

40. Physician's Summary and elaboration of the medical history on front of report. Use additional sheet if more space is needed.

Physical Examination. Give details of abnormal findings under item 20 below.

1. Date of examination		2. Height	3. Weight
4. Blood pressure		5. Pulse	
6. Distant visual acuity uncorrected	right	left	method used
*7. Distant visual acuity corrected	right	left	(data required only if corrective lenses are normally worn)
8. Near visual acuity uncorrected	right	left	method used
*9. Near visual acuity corrected	right	left	(data required only if corrective lenses are normally worn)
10. Color vision			method used
11. Gross visual fields			
12. Hearing	right	left	method used
13. Eyes, general		14. Pupils	
15. Ophthalmoscopic			
16. Ears, general		17. Drums	
18. Heart		19. Vascular system	

20. Details and evaluation of any item 1 through 19, above, reported abnormal and summary evaluation of over-all condition.

21. Did the foregoing examination reveal any mental or physical disability which might cause impaired judgment or motor coordination? ☐ YES ☐ NO

I understand that any of the information in this examination may be used by the Nuclear Regulatory Commission in the exercise of its authority over the licensing of operators.

(Date) _____ (Signature of examining physician)

Typed or printed name of examining physician

Address

State in which licensed

DOCTOR: IT IS REQUIRED THAT EVERY ITEM ON THIS PAGE BE COMPLETED EXCEPT THOSE MARKED WITH * WHEN NOT APPLICABLE.

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CERTIFICATE OF MEDICAL EXAMINATION

Full and accurate completion of NRC Form-396 is essential to the expeditious processing of applications. Incomplete or unclear medical reports will cause delay and requests for further information. When significant abnormalities exist, the detailed reporting of manifestations, diagnoses, and prognoses with supporting data may expedite the processing of applications and eliminate the need for additional correspondence. Special attention is directed to the following items in the Physical Examination section of the report form:

A. Methods, where requested, must be indicated.

B. Visual acuity (Items 6-9) should be reported in a conventional manner. Near-point and far-point designations are not a satisfactory substitute for reports of ACUITY.

C. If corrective lenses are NOT worn, there is no need to complete lines 7 or 9. If corrective lenses are worn, all four items -- 6, 7, 8 and 9 -- must be completed.

D. Visual fields (Item 11) must be reported in degrees or as "full," "complete," or "normal" (or equivalent).

E. If hearing is not normal (Item 12), audiometric data must be reported. (Refer to Item H below.)

F. An indication of "not applicable," "no significant information," or other similar notation should be entered for items where such a notation is appropriate, instead of merely failing to make an entry for such items.

G. Completion of all items, including Item 21, is required, except as indicated in C. above.

H. If color vision (Item 10) or hearing (Item 12) is not normal, results of a practical test should be submitted indicating the degree to which the lack of color vision or diminished hearing interferes with the individual's ability to perform licensed duties.

APPENDIX E

Typical Sample Questions
Operator and Senior Operator Examinations

This appendix contains lists of sample questions typical of those appearing in operator and senior operator written examinations. The questions are grouped within the categories described in this guide, and effort has been made to indicate which questions apply solely to specific reactor types, such as power, test or research reactors.

It should be noted that some appropriate operator or senior operator questions are not listed herein in either the operator or senior operator categories because operator or senior operator duties would not necessitate knowledge of the subject being investigated by the question. For example, at some power reactors, the management has determined that calculations involving determination of control rod worths are sufficiently complex to necessitate that only the plant reactor engineer be responsible for performing the calculations, although the senior operator on shift directs the activities of the licensed operators.

Conversely, some typical senior operator questions given may be investigating areas in which operators at some facilities need a higher level of knowledge because of the nature of their assigned duties; thus, questions of this type might be included in an operator written examination at such a facility. As an example of this aspect, at a low power research reactor, calculations involving control of coolant chemistry are quite simple and straightforward, and operators can and do make such calculations when Management so directs.

Reference will be made to certain types of reactors throughout this appendix. For the purpose of this guide, some definitions of these types are in order for consistency and clarification, and are as follows:

1. POWER REACTORS are those utilized to produce steam primarily in order to drive turbine generators for the production of electricity or providing a means of propulsion. Reactor powers are generally in excess of 100 MW(t).

2. TEST REACTORS are those utilized for materials testing and isotope production; reactor powers are generally in excess of 5 MW(t).

3. RESEARCH AND TRAINING REACTORS -- self-explanatory; reactor powers --large, generally 100 KW(t) - 5 MW(t); small, generally <100 KW(t).

4. CRITICAL FACILITIES are those reactors utilized for critical experiments and training; reactor powers -- generally ≤ KW(t).

I. OPERATOR EXAMINATION

A. PRINCIPLES OF REACTOR OPERATION

In the list that follows, the asterisked questions are types that would usually only appear in examinations given at facilities with reactors operating at power levels greater than 100 KW. Also, questions such as 7 and 8 would only be appropriate for water reactors and reactors with cores containing fertile material, respectively.

1. Briefly explain the following:

   a. Reactor period

   b. Reflector

   c. Reactivity

   d. Criticality

   e. Neutron flux

   f. Nuclear cross-section

   g. Prompt critical

   h. Subcritical multiplication

   *i. Doppler coefficient

2. a. What is the significance of delayed neutrons in reactor control?

   b. What is the origin of these neutrons?

3. a. What is the function of the neutron source installed in the reactor?

   *b. What nuclear characteristics of the reactor core will enable the operator to start up the reactor without this source once considerable high power operation has been logged?

4. *a. What is meant by "xenon poisoning?"

*b. In an operating condition with equilibrium xenon in the reactor, there are certain factors which collectively create this "equilibrium" condition. What are they and how does the magnitude of each change if the reactor power level is quickly lowered and leveled off?

5. As operator, you are bringing the reactor to power on a stable period of 30 seconds. At a certain time the nuclear instrumentation indicates a level of 10 watts. What level will be indicated at the following times:

   a. two minutes?

   b. three minutes?

6. What nuclear phenomenon contributes the largest amount of heat generation:

   a. During the fission process?

   *b. Two hours after the reactor has been shutdown from full power?

7. One frequently hears about "nitrogen-16" associated with water-cooled reactors. What is this, and of what significance is it to the operators?

*8. What effects are primarily responsible for the ability of the reactor core to breed?

8. FEATURES OF FACILITY DESIGN

All of the questions in this list are typical of those that are, in general, appropriate for most reactors that employ a liquid moderator or coolant to surround the core; however, the following comments are appropriate:

1. Questions 3, 6 and 8 would only apply at a power reactor.

2. Schematic detail in Questions 4 and 5 will depend upon reactor size and purpose.

In addition, Question 4 would not be appropriate for a reactor core that utilized only convection cooling and Question 5(b) would not apply to an open pool or tank type reactor facility.

1. Sketch a plan view of the reactor core showing the location of fuel elements, control rods, reflector, irradiation facilities, and neutron detectors.

2. Sketch a fuel element, including the most important dimensions.

3. Describe the flow cycle within the reactor vessel from entry of feedwater until steam is discharged to the turbine.

4. Draw a simplified schematic diagram of the primary coolant system, showing all major components and vital instrumentation. Include flow directions.

5. Sketch flow diagrams for:

   a. The purification system.

   b. The pressure control and relief system.

   Label the major components (pumps, valves, tanks, etc.), show flow directions, and locate all temperature, pressure, flow and level detectors.

6. List the components that are cooled by the intermediate cooling system. Why is such a "buffer" system used instead of river water?

7. Briefly describe the shielding provided at the facility, starting with the reactor core and moving radially outward.

8. a. Describe the reactor coolant pump seal arrangement. Include flow direction, flow rates and where any leak-off goes.

   b. Describe how a seal failure would be detected.

C. GENERAL OPERATING CHARACTERISTICS

This list contains types of questions which pertain to the operating characteristics of specific reactors and auxiliary systems, but are not generally applicable as are those questions listed in Category A. Specifically, Questions 1, 2 and 7 would be most appropriate for power reactor examinations, Questions 4 and 8 primarily appropriate for large research and test reactors, and Question 6 generally would apply only at small research reactors or critical facilities.

Question 3 would only apply at facilities where the reactor moderator or coolant would undergo a considerable temperature swing during operation.

Question 5 is a general type, with only the power level specified in 5(b) being a variable, depending upon the reactor's power range.

1. Describe how the pressurizer functions to limit pressure variations during step load increases and decreases.

2. State whether each of the following occurrences tend to increase or decrease reactivity, assuming that no scrams occur:

a. Feedwater temperature decrease during normal full power operation

b. Turbine trip at full power

c. Loss of power to one recirculation pump -- reactor at full power

3. Does the worth of a control rod change from hot operating conditions to cold operating conditions? Explain your answer.

4. The reactor is started and placed on automatic operation after reaching one megawatt with the regulating rod halfway withdrawn. With no other changes to the reactor or the position of the safety rods, describe the position change of the regulating rod as a function of time.

5. For the operation specified, sketch the traces you would expect to see on a recorder connected to the following nuclear instrumentation channels:

a. Startup Channel during rod withdrawal of one increment followed by a one minute pause; K_{eff} after withdrawal = 0.996.

b. Log N Channel during scram from 20 MW(t). Show trace to 10 minutes following the scram.

6. Assume the reactor is critical at a level of 1 watt. What effect, if any, will be observed if the source is removed at this point?

7. What methods are used to control oxygen concentration in the primary and secondary system and what concentrations are the upper limits during normal reactor operation.

8. What are the fuel cycle reactivity requirements for the following:

a. Cold critical to hot critical

b. Equilibrium xenon

c. Fuel depletion

d. Allowance for xenon override.

D. INSTRUMENTS AND CONTROLS

Many of the questions given for this category; i.e., 1, 2, 5, 6 and 7, may be appropriate for all types of reactors; however, questions 4 and 8 apply only to power reactors, while question 3 again would be a type normally applicable for reactors with power levels greater than 100 KW.

1. Sketch block diagrams of the following nuclear instrumentation channels.

   a. Startup Channel

   b. Log N Channel

   c. Power Range Channel

2. For each nuclear detector in the nuclear instrumentation channels, discuss the following:

   a. Operating principle of the detector

   b. Method used to distinguish neutrons from other radiation.

3. Sketch a compensated ion chamber recorder trace following a scram for the following situations. Start all curves at 100% on the Log N recorder. Indicate on the traces the reasons for all significant slope changes.

   a. CIC Perfect compensation, no power history

   b. CIC Perfect compensation, many MW days operation

   c. CIC Undercompensated, many MW days operation

   d. CIC Grossly overcompensated, no power history.

4. Describe the sequential operation of the automatic plant control system for a load increase demand, starting with turbine throttle valve opening.

5. Briefly describe the operation of the automatic regulating rod control system. A block diagram may be used for clarity.

6. What interlocks must be satisfied before the control rods can be withdrawn for a reactor startup?

7. How is the temperature of the purification water controlled?

8. Discuss in general terms the in-core flux monitoring system, telling why it is needed, what information it gives you, and what protection it affords.

E. SAFETY AND EMERGENCY SYSTEMS

This listing includes eight questions which may appropriately appear in an examination at any type of reactor facility, small or large. Only the nomenclature would differ for given facilities; e.g., the safety amplifier in Question 2 might be another type of electronic tripping device. In addition, the engineered safeguards system referred to in Question 7 and the containment of Question 3 would often not exist at a critical facility or small research reactor.

1. List the parameters, systems or conditions which will produce automatic reactor shutdown and give the set points at which scrams will occur.

2. Explain briefly the operation of the magnet safety amplifier during a scram condition.

3. What signals isolate the containment building?

4. Since the protection system provides scram action, why does it also have a setback action?

5. a. Sketch a simplified schematic diagram and briefly describe the operation of the emergency "poison" system.

   b. What does the "poison" consist of and how great a reactivity increase will it compensate for?

6. List the equipment that is on the circuit supplied by the emergency diesel generator.

7. a. Sketch the Low Pressure Safety Injection system and show major components and valves.

   b. Describe the changes that take place throughout the engineered safeguards systems upon receipt of a Recirculation Actuation Signal.

8. Explain how the possibility of scram system failure is reduced by design.

F. STANDARD AND EMERGENCY OPERATING PROCEDURES

In this listing, only Questions 1.a, 1.b, 4, 6 and 7 are types generally applicable to all classes of reactor facilities. Questions 2, 5 and 8 would be most appropriate for a test reactor or power reactor, while Question 3 is most applicable to a research or test reactor only.

1. List the following provisions of your operating procedures:

   a. Minimum reactor period while increasing power.

   b. Maximum activity of the reactor coolant.

   c. Maximum heating rate for reactor vessel.

2. Briefly outline the MAJOR steps that will be followed in bringing the reactor from full power to cold shutdown.

3. Describe the procedural requirements which must be met before you, as an operator, would insert a sample into or near the reactor for irradiation.

4. Assume that, during a startup, the reactor becomes critical at a significantly lower control rod position than expected. What action should be taken by the operator?

5. While operating at 60 MW(t), reactor power drops suddenly (indicated by ALL power monitors) to 30 MW(t) and begins to level off. What should be your actions as reactor operator on duty? What might have caused this?

6. Assume the reactor is not being operated, (i.e., no control rods withdrawn) but that the control power key is in the switch. What control room personnel requirements must be met?

7. After an extended shutdown, the reactor has just returned to power when an electrical outage occurs. Must the nuclear instrumentation be recalibrated before starting up the reactor? Explain.

8. What immediate action would you take if you received word of a 3 GPM leak in the primary cooling system while you were operating the reactor?

G. RADIATION CONTROL AND SAFETY

All questions listed in this category are general types that are usually appropriate at all types of reactor facilities.

1. Provide values for the following:

   a. The maximum permissible quarterly radiation dose to the whole body.

   b. The maximum quarterly radiation dose allowed for hands.

   c. The maximum quarterly radiation dose allowed for the skin.

   d. The maximum and minimum radiation levels expected in an area marked "Caution, Radiation Area."

   e. The maximum and minimum radiation levels expected in an area marked "Caution, High Radiation Area."

2. Name four types of radiation that could be present in reactor buildings, and list them in decreasing order of biological damage, assuming 1 RAD of each type of radiation.

3. Is there any difference between radiation and contamination? Explain your answer.

4. What is a "Radiation Work Permit?" What items must be listed on such a permit? What conditions terminate an RWP?

5. Tell the following about the Fixed Radiation Monitoring System:

   a. Type of detectors employed.

   b. Types of radiation measured by each of these detectors.

   c. Why is neutron monitoring considered necessary in the spent fuel storage areas?

6. List the portable radiation monitoring equipment available. Indicate the type or types of radiation detected by each instrument.

7. If the radiation intensity is 3.2 Rem/hr at two feet from a small source, what will the intensity be 8 feet from the source?

8. What action would you take after each of the following events occur?

   a. While working with contaminated material, a sharp edge cuts your hand.

   b. While drawing a sample from the purification system, a line parts and you are sprayed with water.

   c. While walking by the Hand and Foot monitor, you notice a high reading on the meter.

H. REACTOR THEORY

In this grouping of typical questions, Nos. 1 thru 4, 7a and 7c would generally be appropriate for all types of reactor facilities, although the answer to Question 3 should contain more explanation for power and test reactors; i.e., to consider the effects of burnup, temperature swing, etc.

Questions 5, 6, 7b, 7d and 8 are usually only appropriate for power or test reactors with Question 5 specifically applicable to a boiling water reactor; however, this type of question could be made appropriate for any power reactor, if slight modifications are made in the terminology used.

1. Assume 1000 neutrons are born from thermal fission. List, in sequence, what can happen to these neutrons. Approximately how many would have to cause fission to sustain a chain reaction?

2. A reactor is operating at a just critical level of 100 watts.

   a. If a 0.002 ΔK step is added, approximately what will the resulting stable period be?

   b. Neglecting the effects of delayed neutrons, approximately what period would be obtained for the step increase in (a)?

3. Discuss how and why the worth of an individual control rod can be variable throughout reactor operation. Include both axial and radial effects in your discussion.

4. A shutdown reactor has a source range reading of 270 CPM. The operator begins to pull one of three supposedly equal worth rods. The count rate rises to 450 CPM when the rod is completely withdrawn. The operator now pulls the second of the three rods and the count rate rises to 1350 CPM when the rod is completely withdrawn. Does the much greater increase in count rate show that the rods are not really equal or is this to be expected? Explain and demonstrate your answer.

5. Discuss how the following factors change, if any, in a boiling water reactor during an increase in power:

   a. Resonance absorption.

   b. Neutron leakage.

   c. Thermal utilization.

6. Compare the reactivity effects of xenon poisoning to that of samarium. Include transient and long-term steady-state effects in your discussion.

7. Define: a. Isotope

b. Burnout heat flux

c. Neutron lifetime

d. Core conversion ratio

8. a. Define the term "nucleate boiling."

b. When this type of boiling occurs, how and why is the heat transfer coefficient affected?

I. RADIOACTIVE MATERIALS HANDLING, DISPOSAL & HAZARDS

In this section, Questions 1, 2, 5, 6, 7 and 8 are generally applicable to all facilities, with the depth of detail for the answers to 1 and 2 depending upon reactor size and purpose.

Question 3 would be primarily appropriate for power or test reactors, while Question 4 illustrates how the same general type of question is appropriate at large research facilities.

1. Sketch a block diagram of the liquid waste disposal system. List the plant drains which go directly to the tanks, and indicate direction of flow and capacity of the tanks. Omit lines below 1 inch diameter.

2. Discuss how gaseous waste is monitored, collected and disposed of at your facility, and list the normal sources of such gases.

3. As senior operator on shift, you are notified that a primary coolant sample has yielded a radioactivity level of 10 times higher than the previously taken sample normal for 20 MW(t) operation:

a. What is the activity level of the high sample?

b. What corrective action, if any, should be taken?

c. How could it be determined whether the sudden increase in activity was due to a fuel element rupture or a "crud burst?"

4. With the reactor operating at 1 MW(t), one of the fuel elements ruptures.

   a. How would the oeprator first receive an indication of the failure?

   b. As senior operator on duty, what should be your immediate actions upon suspecting the element has failed?

   c. Outline the methods that should be used to remove the element and clean up the pool.

5. What is the approximate attenuation to a 1 Mev gamma rays afforded by:

   a. One inch of lead shielding?

   b. One foot of water shielding?

      μ of lead = 2.3/inch

      μ of water = 0.19/inch.

6. An operator in his haste to run from a suddently exposed gamma source, trips and strikes his head. Assuming that he lay unconscious three feet from the source, calculate the following:

   a. Average whole body dose received in one hour.

   b. If the man were not removed for eight hours, discuss the probable state of his health after the exposure.

   Note: Source strength is 20 curies, emitting gamma energies of 1.7 and 1.4 MEV.

7. What factors should be considered by the shift supervisor when he assigns jobs that involve radiation exposure?

8. Assume there is a release of radioactive material within the reactor building and it is necessary that your operators enter the building area without waiting for a health physicist to arrive.

   a. What kind of monitoring equipment would you require?

   b. What precautions would you take prior to entry?

   c. Explain in detail how you would determine how long the operators could stay in the vessel.

J. SPECIFIC OPERATING CHARACTERISTICS

Typical questions similar to Nos. 4, 5, and 7 would be appropriate for all reactor facilities.

Questions 2 and 8 generally are only appropriate for power reactors, although Question 8 could be modified for use on a large research or test reactor examination.

Question 3 is specifically applicable to a pressurized water reactor.

Questions 6 and 7(b) are appropriate for either power, test, or large research reactors.

As stated in Part V of the guide, for typical questions involving interpretation and use of graphs or curves, the curves would be distributed along with the examination. Thus, the appropriate curves would accompany Question 8.

1. Sketch graphs that show the response with time of the following variables following a turbine trip. Assume the plant was at 100% power on fully automatic control and that no operator action is taken following the trip:

   a. Reactor power.

   b. Reactor outlet temperature.

   c. Reactor inlet temperature.

   d. Steam outlet temperature.

   e. Steam flow (per loop).

2. On your answer sheet, list the following ranges of values for normal power operation, including units, in tabular form.

	Primary Coolant	Intermediate Coolant
Max. Total solids	_____	_____
Max. dissolved solids	_____	_____
Max. chloride	_____	_____
Dissolved hydrogen	_____	_____
pH	_____	_____

3. The reactor power is 70%; Group D control rods are at 210 steps and in manual; the boron concentration is 510 ppm. This power level has been maintained for two days.

   a. Calculate the quantity (in gallons) of water and/or boric acid necessary to increase the power level to 100%. Show all work and mark all curves to indicate where you got your information (See Figures J1 - J9).

   b. The power is to remain at 100% for some time. Calculate the ppm change in boron necessary to maintain proper operation over the next 2 days.

   c. Describe how the operator will know when and how fast to add water and/or boric acid to maintain proper operation.

   d. How and why will the axial flux imbalance change when going from 70% to 100% power?

4. The reactor is critical at 50 watts of power and increasing on a stable reactor period. The startup channels show a 100 second period and the intermediate channels show a 65 second period. Which period do you consider is the more accurate? Explain your answer.

5. Outline a method that may be used to determine the cold minimum core shutdown margin during low reactor power operation.

6. When the axial flux shape is plotted after one day's operation at full power and again after 21 days full power operation, the maximum flux level and location is different for the two cases.

   a. Will the maximum flux be located higher or lower in the core at the later date?

   b. Explain your answer.

7. a. How are the power range nuclear detectors calibrated?

   b. Using typical full power values for the parameters needed, calculate the reactor thermal power.

8. Assume the reactor has been at full power, 600 MW(t), for two months. Controlling Rod Group "A" is banked at 27". A full scram occurs, the cause of the scram has been determined, the situation remedied and the control rods are to be withdrawn. Calculate the just critical control rod bank position necessary to reach a power level adequate to commence nuclear heating.

K. FUEL HANDLING & CORE PARAMETERS

In this grouping, Questions 2, 4(a), 5, 6, and 7 are appropriate for all reactor facilities, except that the size and amount of fuel handling equipment referred to in Question 6 will be quite variable, e.g., at a small research reactor, the fuel handling apparatus may consist of a hook tool and a rope snare.

Questions 1, 3, 4(b) and 8 apply primarily to power reactors, with possible application of 4(b) to large research and test reactors.

1. During fuel loading the records become confused so that the enrichment of a new fuel element is not known for sure. Where and what type of identification is on the fuel element to indicate the enrichment of the element?

2. As the water to metal ratio of a reactor core is increased over that which is optimum, reactivity decreases due to an increasing "overmoderation" condition. Explain WHY this phenomenon causes a reactivity decrease, past the optimum point.

3. During a normal plant heatup from 250°F, the reactor is maintained at 1×10^{-7} amps on channel 5 to produce a heatup rate of 35°F/hr. Will the same indicated power produce the same heatup rate as the plant approaches operating condition? Explain your answer.

4. a. List and give the half-lives of installed sources in the reactor core, and explain how each source produces neutrons.

   b. Explain how the nuclear characteristics of the Antimony-Beryllium source enable it to be termed "regenerative".

5. A startup detector is placed near a reactor core to monitor a new core loading. As the loading proceeds, the following data is obtained.

Step **	No. of Elements	CPS **
	0	10
1	4	12
2	8	15
3	10	20
4	12	30
5	13	60
6	14	150

a. Estimate the number of fuel elements that will make the reactor critical and show how you arrived at this figure.

b. Do you think the detector is optimumly placed, too close or too far away from the source? Explain.

6. List the major equipment which comprises the fuel handling system, and briefly describe the purpose of each.

7. What procedural requirements must be met before control rod or fuel followers may be transferred to or from the core?

8. a. What is the purpose of control rod programming?

b. Explain why the fuel enrichments in certain sections of the core differ from each other.

L. ADMINISTRATIVE PROCEDURES, CONDITIONS & LIMITATIONS

In this section, Questions 4, 5, 6 and 8 are usually appropriate for all reactor facilities.

Questions 1, 2 and 3 are generally applicable for power reactors, while Question 7 would be most appropriate for research and test reactors.

1. There are 3 types of tags used at your facility: Danger, Caution and Hold For Inspection.

a. Indicate when each type of tag would be used.

b. What is the procedure for "tagging clearance"?

2. List the following administrative limitations:

a. Maximum rate of generator load increase or decrease.

b. Containment building maximum internal pressure.

c. Maximum reactor power with vessel head removed.

3. Assume that you are shift supervisor and no higher supervision is at the plant. The reactor is operating at full power and a scram of undetermined origin occurs. The system dispatcher informs you that the power is badly needed on the line. How long would you wait before ordering the reactor to be started up?

4. Which individuals, by position title, in the plant organization may authorize the following operations?

a. Adjusting the discrimination in the startup channel.

b. Disposing of liquid waste from the site.

c. Scramming the reactor.

5. Assume that you are the senior operator on duty and are supervising a core loading change. An operator is lowering a fuel element into the core (it is halfway inserted) when the reactor top fixed radiation monitor alarms. What should be your actions?

6. Under what conditions must a proposed change to a reactor operating procedure be referred to the Safeguards Committee for review?

7. List the types of experiments which will NOT be accepted for insertion in the core region, according to the technical specifications.

8. On your answer sheet, complete the following table by listing the personnel that MUST be in each status for the event given.

Event	In Control Room	At Facility	On Call
a. Replacement of fuel in core.----------	_____	_____	____
b. Reactor re-start after scram from high flux.--	_____	_____	____
c. Reactor power escalation after a 10% power drop, cause unknown. Conditions appear normal.----------	_____	_____	____
d. Normal reactor shutdown.----------	_____	_____	____

APPENDIX F

Eligibility for Examination With No Reactor Startup Demonstration

Normally, a reactor startup is required as part of the operating test administered at the facility. However, an alternative program, described below, may be completed for power reactor applicants in which case no actual reactor startup needs to be performed during the license examination.

REQUIREMENTS TO BECOME ELIGIBLE FOR EXAMINATION AT AN OPERATING POWER REACTOR WITH NO REACTOR STARTUP DEMONSTRATION.

A. Educational Requirements

As specified in ANSI N18.1-1971.

B. Lectures

A total of 500 hours on subjects listed in ANSI N18.1-1971, Section 5.2.1, related subjects and prerequisite courses. These lectures are in addition to the lectures in the one week training program in Section D below.

C. Power Plant Experience

Applicants shall meet the following requirements:

1. ANSI N18.1-1971, Section 4.5.1 or 4.3.1 as applicable.

2. A minimum time of six months at the site. At least three months of this time shall include participation in on-the-job training which involves manipulation of the nuclear power plant controls during day to day operation. This phase of the training will be programmed and supervised. A record of training accomplishment should be maintained and signed off by the training coordinator or other responsible individual.

3. Applicants must participate in reactor and plant operation at facility power levels up to at least 20% power prior to the examination.

4. The applicant has manipulated the controls of the reactor facility during five significant reactivity changes as described in the operator requalification program for the facility. Every effort should be made to have a diversification of reactivity changes for each applicant.

D. Simulator Training

The applicant has satisfactorily completed a NRC approved training program of at least one week duration at a nuclear power plant simulator. The application shall contain a certification from the simulator training center attesting to the applicant's:

1. ability to manipulate the controls and keep the reactor under control during a reactor startup,

2. ability to predict instrument response and use the instrumentation during a reactor startup,

3. ability to follow the facility startup procedures, and

4. ability to explain alarms and annunciators that may occur during this operation.

E. Review and Evaluation

A minimum of 40 hours for review, audit examinations and evaluation of the applicant should conclude the training program.

F. Records

Documentation shall be maintained as specified in ANSI N18.1-1971, Section 5.6.

APPENDIX G

NRC FORM 157 (10-73) — U.S. NUCLEAR REGULATORY COMMISSION

LICENSE APPLIED FOR: RO ☐ SRO ☐

CURRENTLY A LICENSED OPERATOR HERE? YES ☐ NO ☐

EXAMINATION REPORT

APPLICANT'S NAME	REACTOR	LOCATION

WRITTEN EXAMINATION

OPERATOR (SECTION 55.21)	ADMINISTERED BY:	DATE	WAIVED ☐
	GRADED BY:	GRADE %	EVALUATION PASSED ☐
	CATEGORY GRADES A____% B____% C___% D____% E____% F____% G___%		FAILED ☐
SENIOR OPERATOR (SECTION 55.22)	ADMINISTERED BY:	DATE	WAIVED ☐
	GRADED BY:	GRADE %	EVALUATION PASSED ☐
	CATEGORY GRADES H____% I____% J____% K____% L____%		FAILED ☐

OPERATING TEST

(SECTION 55.23)	ADMINISTERED BY:	DATE	WAIVED ☐ PASSED ☐ FAILED ☐
			HOT ☐ COLD ☐

COMMENTS

RECOMMENDATION	APPROVE FOR SENIOR LICENSE ☐ OPERATOR LICENSE ☐	SIGNATURE
	DO NOT APPROVE FOR SENIOR LICENSE ☐ OPERATOR LICENSE ☐	EXAMINER DATE
ISSUE LICENSE	SENIOR LICENSE ☐ OPERATOR LICENSE ☐	SIGNATURE Chief, Operator Licensing Branch

NOTE: The operator and oral examination check sheet must be completed for each applicant. If applicant has failed the operating and oral test at the Senior Level, the check sheet should be completed to indicate applicant's evaluation at the operator level.

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OPERATING AND ORAL EXAMINATION SUMMARY REPORT

If an SRO applicant is unsatisfactory at the SRO Level, complete the RO column to evaluate applicant's adequacy at the RO Level. **For each unsatisfactory ("U"), list the page number(s) of the operating oral examination notes on which the unsatisfactory responses are explained.*	EVALUATION		
	SRO	RO	*PAGE NUMBER FOR "U"
1. OPERATING DEMONSTRATION			
1.1 Pre-startup and Instrument Checks			
1.2 Console Operation			
a. Manipulations			
b. Understanding			
2. FACILITY EQUIPMENT			
a. Major			
b. Auxiliary			
c. Engineered Safeguards Systems			
3. INSTRUMENTATION			
a. Nuclear			
b. Process			
4. REACTOR PROTECTION			
5. PROCEDURES			
a. Normal			
b. Abnormal			
c. Emergency			
6. REACTIVITY EFFECTS (Except Console Operation)			
7. ADMINISTRATIVE REQUIREMENTS			
8. EMERGENCY PLAN			
9. RADIATION PROTECTIO			
10. RESPONSIBILITIES			
COMMENTS			

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OPERATING AND ORAL EXAMINATION NOTES

A. OPERATING DEMONSTRATION	EVALUATION
1.1 Pre-startup and Instrument Checks	
Type of checkout: *(specify)* _____	
1.1.1 Familiarity with check sheet	
1.1.2 Accuracy when reading instruments	
1.1.3 Understanding of what is being checked	
1.1.4 Understanding of reasons for checkout	
1.1.5 Effects of malfunctions	
1.1.6 Knowledge of Control Room reference data	
1.1.7 Method of ECP determination	
1.2 Console Operation	
a. Initial conditions:	
b. Program:	
1.2.1 Follows procedures	
1.2.2 Ability to predict response for specified program	
1.2.3 Observes and checks instrumentation	
1.2.4 Understanding of instrument response	
1.2.5 Ability to follow specified program accurately	
1.2.6 Dexterity and "feel" for console controls	
1.2.7 Knowledge of reactivity effects	

COMMENTS: *(Required for "U")*

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B. CONTROL ROOM *(Major, Auxiliary and Engineered Safeguards Systems)*	SYSTEMS							
	A	B	C	D	E	F	G	H
2.0 EQUIPMENT								
2.1 Purpose								
2.2 Flow Path								
2.3 Normal Parameters								
2.4 Components								
2.5 System Behavior and Response								
3.0 INSTRUMENTATION								
3.1 Detector								
3.2 Malfunction								
3.3 Control Room Indication								
4.0 REACTOR PROTECTION								
4.1 Alarms/Setpoints								
4.2 Safety System Input								
4.3 Interlocks								
5.0 PROCEDURES								
5.1 Normal Procedures								
5.2 Abnormal Procedures								
5.3 Emergency Procedures								
6.0 REACTIVITY EFFECTS								
7.0 ADMINISTRATIVE REQUIREMENTS								
7.1 Technical Specifications								
7.2 Facility Requirements								

COMMENTS: *(Required for "U")*

☐ CONTINUED ON REVERSE

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8. CONTROL ROOM *(Nuclear and Radiation Instruments)*	SYSTEMS A	B	C	D	E
3.0 INSTRUMENTS					
3.1 Detectors					
3.2 Malfunctions					
3.3 Control Room Indications					
3.4 Channel Components					
3.5 Compensation/Discriminator					
3.6 Input to Control System					
4.0 REACTOR PROTECTION					
4.1 Alarms/Setpoints					
4.2 Safety System Input					
4.3 Interlocks					
5.0 PROCEDURES					
5.1 Normal Procedures					
5.2 Abnormal Procedures					
5.3 Emergency Procedure					
7.0 ADMINISTRATIVE REQUIREMENTS					
7.1 Technical Specifications					
7.2 Facility Requirements					

COMMENTS: *(Required for "U")*

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B. CONTROL ROOM *(Electrical)*	SYSTEMS A	B	C	D
2.0 EQUIPMENT				
2.1 Purpose				
2.2 Flow Path				
2.3 Normal Parameters				
2.4 Components				
2.5 System Behavior or Response				
3.0 INSTRUMENTS				
3.2 Interlocks				
3.4 Control Room Indication				
5.0 PROCEDURES				
5.1 Normal Procedures				
5.2 Abnormal Procedures				
5.3 Emergency Procedures				
7.0 ADMINISTRATIVE REQUIREMENTS				
7.1 Technical Specifications				
7.2 Facility Requirements				

COMMENTS: *(Required for "U")*

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C. REACTOR AND AUXILIARY BUILDINGS
(Major, Auxiliary, Electrical Safeguards, Fuel Handling, Rad Waste)

		SYSTEMS					
		A	B	C	D	E	F
2.0	EQUIPMENT						
2.2	Flow Paths						
2.3	Normal Parameters						
2.4	Equipment Location						
2.5	System Behavior and Response						
3.0	INSTRUMENTS						
3.8	Local Instrumentation						
5.0	PROCEDURES						
5.1	Normal Procedures (Local)						
5.2	Abnormal Procedures (Local)						
5.3	Emergency Procedures (Local)						
6.0	REACTIVITY EFFECTS						
7.0	ADMINISTRATIVE REQUIREMENTS						
7.1	Technical Specifications						
7.2	Facility Requirements						
9.0	RADIATION PROTECTION						
9.1	Radiation Control Procedures						
9.2	Portable Instruments – Location						
9.3	Portable Instruments – Use						
9.4	Portable Instruments – Characteristics						
9.5	Personal Practices						
8.0	EMERGENCY PLAN						
8.1	Equipment Locations						
8.2	Equipment Use						
8.3	Duties						

COMMENTS *(Required for "U")*

NRC FORM 157
(10-75)

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D. DISCUSSIONS *(Integrated Plant Response)*		SYSTEMS			
		A	B	C	D
2.0	EQUIPMENT				
2.8	Components Response				
3.0	INSTRUMENTS				
3.4	Control Room Indications				
3.8	Automatic Control				
3.9	Ability to Manipulate Manual Control				
4.0	REACTOR PROTECTION				
4.1	Automatic Actions				
4.2	Alarm/Setpoints				
5.0	PROCEDURES				
5.1	Normal Procedures				
5.2	Abnormal Procedures				
5.3	Emergency Procedures				
6.0	REACTIVITY EFFECTS				
6.3	Coefficient Effects				
6.8	Transient Analysis				
7.0	ADMINISTRATIVE REQUIREMENTS				
7.1	Technical Specifications				
7.2	Facility Requirements				

COMMENTS *(Required for "U")*

NRC FORM 157
(10-73)

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D. DISCUSSION	EVAL-UATION
8.0 REACTIVITY EFFECTS *(Nuclear Theory)*	
8.1 Subcritical Multiplication	
8.2 Delayed Neutrons Effect	
8.3 Coefficients	
8.4 Poison Effects	
8.5 Long Term Exposure Effects	
9.0 RADIATION PROTECTION	
9.1 Radiation Control Procedures	
9.4 Source and Types of Radiation	
9.6 Contamination	
9.7 Exposure Limits *(10 CFR 20) (Facility)*	
10.0 RESPONSIBILITY (QA, NON-RAD SAFETY, AP's, SECURITY)	

COMMENTS *(Required for "U")*

NRC FORM 157
(10-75)

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1. Identify the law which gives the Nuclear Regulatory Commission authority to regulate operator licensing:

2. List three possible exceptions to the general rule that no one can manipulate the controls of a nuclear reactor without a license granted by NRC:

3. List and explain two broad areas of inquiry contained in the license application:

4. List three elements of 10CFR55.11 that must be satisfied before a license is granted:

5. List eleven areas covered by an operator's written examination:

6. List nine additional areas covered by a senior operator's examination:

7. List five basic limitations on an operator license:

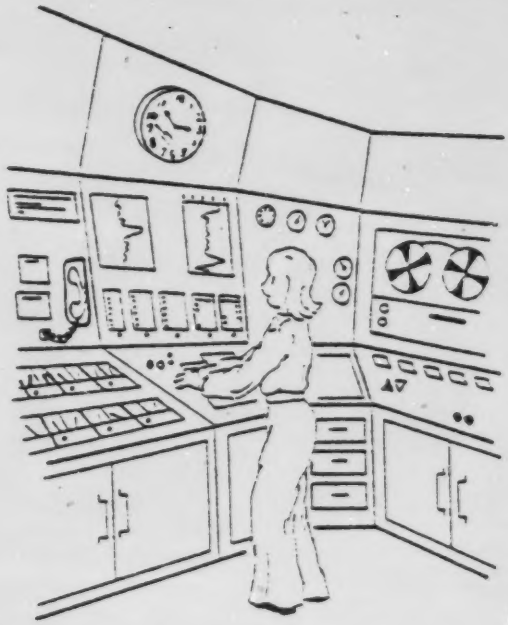
8. State the general duration of a license:

9. List five conditions required for renewal:

NUCLEAR TECHNOLOGY

MODULE 7-2 ADMINISTRATIVE CONTROL FOR NUCLEAR SYSTEMS. -

JAMES R. PENDLAND



REACTOR OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER - SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 7 REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

By now the student is sure to be aware that the operation of a nuclear power plant involves many complex systems, with each system operated according to well defined and strictly performed procedures. In addition, design, construction and maintenance are subject to strict compliance with written procedures. These procedures and the licensing process make up the general administrative controls for nuclear power systems.

The prospective operator of a reactor system must be familiar with these administrative controls because they directly affect virtually every action taken in the performance of an operator's duties. In fact, the administrative controls actually dictate the philosophy by which the operations are performed.

This module acquaints the student with the history of regulation of nuclear power, from the Atomic Energy Commission (AEC) through the Reorganization Act of 1974, which split the AEC functions and distributed them between the Nuclear Regulatory Commission (NRC) and the Energy Research and Development Administration (ERDA), to the Department of Energy (DOE) Organization Act of 1977, which joined all Federal energy agencies, except NRC, into one department.

Also, the licensing process is outlined, from the initial application for a construction permit, through the NRC's granting of an operating license, including intermediate steps such as the Preliminary Safety Analysis Report (PSAR), the Environmental Impact Statement and the Final Safety Report (FSR).

Finally, the utility organization is discussed in general terms to give the student an idea of how administrative controls apply to a licensed operator.

MODULE PREREQUISITES

The student should have completed Course 5 in this series, "Introduction to Nuclear Systems."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Name two Federal entities resulting from the Organization Act of 1977; outline responsibilities of each.
2. Identify purpose of construction permit; briefly describe its contents as required by 10CFR50.34.
3. Briefly outline criteria for issuance of a construction permit.
4. Distinguish between a Preliminary Safety Analysis Report and a Final Safety Report.
5. Describe purpose of an Environmental Impact Statement; list four possible areas of investigation for report.
6. List general criteria for issuance of an operating license.
7. Describe double level of enforcement exercised by NRC.
8. Draw a basic utility company organizational chart.

HISTORY OF NUCLEAR REGULATION

Prior to World War II, pioneers of the nuclear frontier were virtually unregulated in their endeavors. Even when the war effort resulted in creation of the Manhattan Project, there was little to be regulated except for top secret bomb development. As the war ended and interest in the power of the atom spread, the U.S. Government quickly set up an agency that could safeguard military advances in the field of nuclear weapons and, at the same time, develop peaceful uses of the atom.

As a result, the AEC was established (Figure 7-2-1), with two branches that answered to the commissioners. These two branches, though equal in status, served completely different functions.

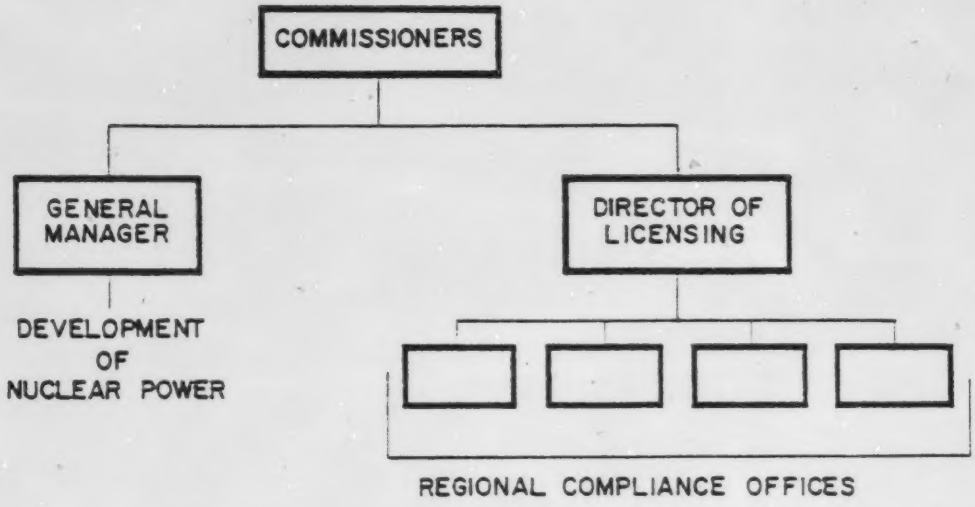


Figure 7-2-1. Organization of AEC prior to Reorganization Act of 1974.

The General Manager, from the office in Germantown, Pennsylvania, was charged with development of nuclear power. Fuel processing, reactor design development and promotion of good public attitude about the new source of power were included in the responsibility. On the other hand, Bethesda, Maryland, headquartered the Director of Regulation, who controlled actions of the regional officers. Their jobs were to ensure that rules and regulations of the Commission were strictly enforced.

In 1974, Congress passed the Energy Reorganization Act, which separated development from regulation, with the goal of ensuring that the agency doing the regulating was not regulating itself and, therefore, biased in its method or degree of regulation.

To meet this goal, the job of promoting and developing nuclear power was assigned to ERDA (now a part of DOE), and NRC was charged with ensuring that nuclear activities within the U.S. were performed in compliance with rigid national regulations and standards.

Because this module is concerned with administrative controls on nuclear power plants, its focus is on activities of the NRC. From its headquarters in Bethesda, Maryland, the commissioners set policy and administer in such areas as plant and plant operator licensing; however, in the area of compliance, agents from regional offices throughout the nation (from which they travel to the individual facilities) make the inspections.

These inspectors arrive at a facility with or without prior notice and check to make sure the plant is being operated according to specifications set out by NRC.

But, before dealing further with these inspections, a discussion of the process that leads from the application

for a construction permit to the operational startup of the plant and administrative controls involved in this process is presented below.

LICENSING PROCESS

The first step toward construction of a nuclear power plant is to obtain a construction permit from NRC (hereafter referred to as the Commission).

CONSTRUCTION PERMIT

To facilitate description of the licensing process, assume a fictional utility company known as Acme Power and Light. Further assume that Acme has decided to build a new nuclear power plant to increase its output capacity.

Acme is not permitted to begin construction on the plant until certain actions are taken and approved by the Commission (and various other local, state and Federal agencies). The first requirement is for submission (for approval) of an application for a construction permit - permission to build the plant.

Application

The application must be submitted to the Commission well in advance of the time construction is to begin because approval is not given without serious and detailed study by the Commission. The permit procedure affords the Com-

mission a good opportunity to head off problems prior to any unnecessary expenditure of time and money by Acme. In short, the Commission examines Acme's proposal (application) and then tells the company whether or not it is acceptable.

Contents of the Application

Title 10, Part 50.34 of the Code of Federal Regulations (CFR) sets out minimum requirements for the contents of a construction permit application. The idea of the application is to tell the Commission what Acme intends to build; therefore, it must begin with a summary description of the type of power plant Acme wants, including such items as kind of reactor (BWR or PWR), the kinds of safety features to be incorporated to protect the reactor from power excursions or prevent the release of radioactivity, and any planned innovations in plant design. Furthermore, the application must include a description of the site upon which the facility is to be situated, as well as any design considerations that might stem from the siting.

Acme must include a description of a proposed training program for plant operators. Also, the plan to be used to cope with emergencies that might arise during operation of the reactor must be presented in detail. In summary, the application describes for the Commission what Acme plans to do.

Issuance of the Construction Permit

Once the Commission is satisfied that construction is in the best interest, it can authorize construction of the plant.

(Note: Issuance of a construction permit does not

assure the utility of an operating license when construction is completed. Before the Commission will issue an operating license, Acme must complete construction according to the permit and submit a Final Safety Report. If the Commission finds that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the requirements of the license and regulations as set forth by the Commission, it may then issue the operating license.)

PRELIMINARY AND FINAL SAFETY REPORTS

As stated above, the licensing process requires that both a Preliminary (PSAR) and Final (FSR) Safety Report be submitted to the Commission before Acme can be awarded an operating license. As may be apparent from the names of the reports, the former (PSAR) provides as much information about the proposed facility as is available at the time. The second report (FSR) contains much of the general information previously presented in the PSAR, but must go further, adding (as a minimum) the following:

- Current information on items under investigation during construction period, such as environmental or other measures.
- Final evaluation of the design of systems and components within the systems.
- Applicant's organizational structure, and administrative controls applied by Acme to operation of the facility (if license is issued).
- Plans for preoperational testing of the facility.

Plans for normal and emergency operation of the facility.

Plans for coping with emergencies that may arise, whether from the facility itself or from outside the facility.

Technical Specifications, as set forth on 10CFR50.36. (Technical Specifications include items in the following categories: safety limits, limiting safety system settings and limiting control settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls).

Acme's qualifications to operate the facility within the standards set out by the Commission . . .

> The requirement that the company be qualified is based on the same idea as the requirement that the operator be qualified to perform his or her duties. The Commission is not allowed to permit operation of a complex nuclear power plant by a company that has neither the technical facilities nor the financial stability to adequately ensure safe operation of the plant over its lifetime.

The FSR provides information not provided in the PSAR, and it ties up any loose ends in the task of ensuring that the facility is built in such a way that the public health and safety are reasonably protected.

ENVIRONMENTAL IMPACT STATEMENT

In recent years it has become apparent that nearly everything society does, either constructive or otherwise,

has an effect on the environment and quality of life. For this reason, regulatory bodies of the Federal and state governments require that a study be done prior to major construction activities to determine to what extent the environment will be affected and what devices can be employed to minimize this effect. While nuclear power plants are not the only facilities required to submit such environmental studies, they can affect the environment in a variety of ways, and must submit an Environmental Impact Statement.

This statement, submitted with the application for a construction permit, must include results of an investigation to determine:

Effect of construction on wildlife and vegetation in areas adjacent to the construction site.

Effect of possible introduction of warmer water into areas where fish and vegetation exist prior to construction.

Effect of sanitary discharges from the plant as a consequence of human inhabitation of previously uninhabited areas.

Potential release of radioactivity or chemical components as a result of plant operation.

An analysis of cost versus benefit of plant operation.

The primary purpose of the Environmental Impact Statement is to ensure that the environment is taken into consideration before a major construction job is begun. A decision on the environmental impact must be based on a balance of the need for additional power generation capacity against the effect the plant will have on the environment. In the end, it comes down to a case of setting priorities.

GENERAL REQUIREMENTS FOR ISSUANCE OF A LICENSE

Once the Commission is satisfied that Acme has filed all the required applications and statements and is qualified to perform according to the limits of the license and other rules and regulations to which Acme will be subject, a license may be granted. Even then, however, it is issued on the condition that the licensee is causing no unreasonable risk to public health and safety. If Acme intends to keep its license, it must demonstrate on a continuing basis that operation is in accordance with the rules and regulations set forth by the Commission in the license. In other words, granting of the license does not relieve Acme of its responsibilities - it must comply or suffer the consequences. And the consequences may involve a suspension or even a loss of the license.

COMPLIANCE AND ENFORCEMENT OF LICENSE REQUIREMENTS

The Nuclear Regulatory Commission invokes a double-level enforcement procedure that includes external surveillance by the Commission and internal policing by the utility of its own activity. The Commission, as stated previously, may check Acme for compliance with operating requirements and, if items of noncompliance are found, take appropriate action to ensure compliance. Such action may consist of a fine, suspension or revocation of the license.

The Commission also requires that Acme report any items of noncompliance; the reporting requirements vary with kinds and seriousness of the noncompliances.

UTILITY ORGANIZATION AND ADMINISTRATIVE CONTROL

The Commission does not prescribe the organizational structure for any utility company. A typical organization (Figure 7-2-2) might include a plant manager who supervises

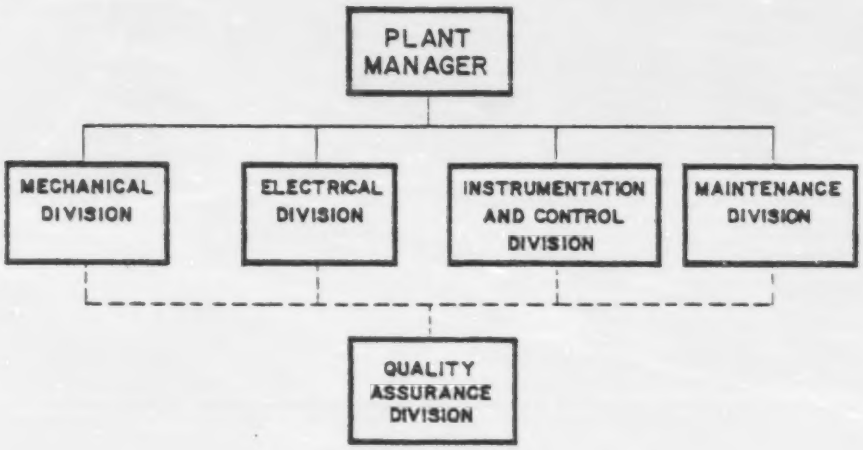


Figure 7-2-2. Typical nuclear facility organization.

the activities of the department managers. Under each of these managers there may be supervisors; for example in the operations division, the mechanical, electrical and instrumentation and control department supervisors ensure proper operation in their specialty and then report to the operations manager. By having the system set up in this manner, the plant manager can ensure compliance with requirements of the license, internal procedures through delegation of authority, and a system that reports the status of departments on a continuing basis.

Note: In Figure 7-2-2, the Quality Assurance Division serves as an independent monitor of performance in the other divisions. Again, the purpose is to ensure compliance with the numerous regulations and procedures set forth by the Commission.

CONCLUSION

A utility company must deal with many administrative controls, from the time it decides to build a nuclear power plant until the plant is decommissioned. Because of this situation, an operator performs every task under administrative control and must appreciate the force of these controls.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

U.S. Nuclear Regulatory Commission. "Requirements for the Physical Protection of Nuclear Power Reactors." Code of Federal Regulations, Title 10 - Atomic Energy, Federal Register 42, No. 37, February 24, 1977.

Available by writing Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

1. Name the Federal agency that regulates nuclear reactor installations and operations:

2. List contents of a nuclear reactor construction permit:
   a.
   b.
   c.
   d.
   e.
   f.
   g.
   h.

3. In one sentence, define criteria by which NRC issues a construction permit:

4. Briefly describe the following:
   a. Contents of PSAR.

   b. Contents of FSR.

5. a. Briefly state purpose of an Environmental Impact Statement:

b. List four items covered by an Environmental Impact Statement:
(1)
(2)
(3)
(4)

6. In general, identify criteria by which NRC issues an operating license to a utility:

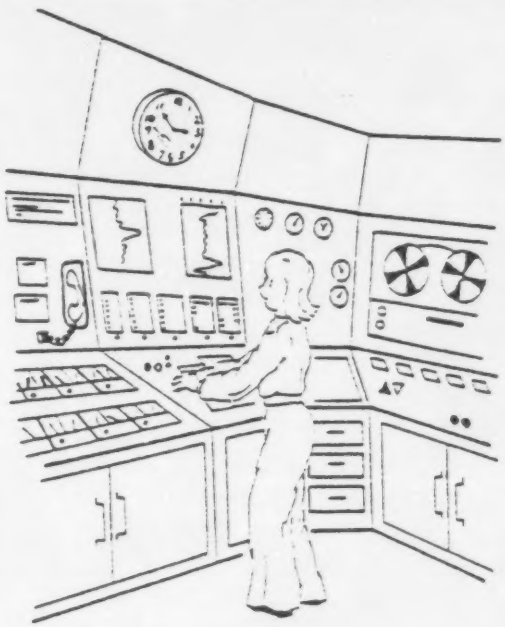
7. Name two levels of enforcement exercised by NRC:
(1)

(2)

8. Draw a typical utility organization chart:

NUCLEAR TECHNOLOGY

MODULE 7-3 REACTOR OPERATING EXPERIENCES



REACTOR
OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER – SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 7 REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

Significant progress has been made in the 35 years since the first nuclear reactor attained criticality under a Chicago stadium. The United States presently has dozens of nuclear power plants with gross electrical power output of more than 100 MWe. The total number of operational plants times their years in operation provides us with at least 300 plant years of experience. (This experience does not include the many research and training reactors located at major universities throughout the country.)

A major contributor to the operational experience gained thus far has been that of reactor safety. A review of the experience shows that the record of nuclear reactors to date has been very good. There have been no accidents harmful to public health and safety. In decades of operation only seven fatalities have been directly related to nuclear accidents. Four resulted from accidentally critical assemblies and three followed a single reactor accident.

When considering that reactor accident, it is usually assumed that the reactor was operating at power; ironically, however, it occurred while the reactor was shutdown and depressurized. It should be noted that there have been no accidents involving reactors operating routinely at normal power.

Reactor operating experience has shown that nuclear reactors are both safe and reliable. Because of the size and complexity of operating plants, however, incidents do occur. Experience has resulted in elimination of designs such as the one involved in the SL-1 accident. Also, experience has resulted in recognition of problems before they evolve into incidents; and incidents at one operating plant provide data for incident elimination or reduction of possibilities

at other operating nuclear plants. This reactor operating experience, coupled with continued reactor safety, has led to increased safety and increased availability of nuclear power.

MODULE PREREQUISITES

The student must have a basic understanding of reactor operations and design, reactor safety, instrumentation and control systems, radiation protection and basic radiation physics, as might be obtained by completion of Courses 1, "Radiation Physics," 2, "Radiation Protection," 5, "Introduction to Nuclear Systems," 6, "Instrumentation and Control of Reactors and Plant," and 8, "Reactor Safety." In addition, the student must have a general understanding of reactor operator licensing and administrative controls, as would be obtained through satisfactory completion of Module 7-1, "Reactor Operator Licenses" and Module 7-2, "Administrative Control for Nuclear Systems."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe four actual reactor accidents, giving descrip-

tion, consequences, reasons, and what provisions have been taken to ensure that these accidents will not reoccur in U.S. commercial nuclear power plants.

2. Describe at least two other criticality accidents, giving description, consequences, reasons, and what provisions have been taken to ensure that these accidents will not reoccur in U.S. commercial nuclear power plants.
3. Describe generic design problems (include fuel densification, tube denting in steam generators, main steam relief system failures, main condenser tube failures, and BWR primary piping cracks); give description, actual and potential consequences and corrective actions.
4. Discuss various types of reactor incidents resulting in forced outages and/or violation of Technical Specifications; include description, actual and potential consequences and corrective actions.
5. Discuss reliability and availability of nuclear power plants; give percentages of different outage causes and compare with large fossil units.

REACTOR ACCIDENTS

The operating experience safety record gained to date, and the safety and availability of present U.S. commercial power plants, have partly resulted from lessons learned from reactor accidents that have occurred. This is not to say that accidents are the only learning mechanism, but to emphasize that as a result of these accidents much has been done to prevent their reoccurrence. Knowledge gained from these accidents has contributed to the remarkable safety record of present plants. To provide a basic understanding of the information obtained from an accident, four major accidents are discussed.

THE SL-1 INCIDENT

The first power reactor accident to result in human fatalities occurred on January 3, 1961, at the Stationary Low Power Reactor No. 1, designated SL-1. This reactor, located at the Atomic Energy Commission's (AEC) National Reactor Testing Station, Idaho Falls, Idaho, was a natural recirculation boiling water reactor (BWR) designed to operate at a thermal power of 3 megawatts (MW). Electrical output of the facility was approximately 200 thousand watts (200 kW), and a little over a million (10^6) Btu/hr output was used for space heating.

The accident, which resulted in the loss of three lives, occurred on the "four-to-midnight" shift. The three reactor operators were assigned no more than a supposedly routine maintenance operation on a shutdown reactor. The first

indication of any problem or trouble at the SL-1 reactor was an alarm received at the AEC fire station and security headquarters. This alarm occurred at approximately 9:00 o'clock on the evening of Jan. 3, 1961. At the same time, a personnel radiation monitor about a mile from the site alarmed and remained unsteady for several minutes.

After the alarm was received, both AEC fire department and security forces responded to the SL-1 site. Entry was made to the site by firemen with self-contained breathing units and radiation survey instruments. They entered the administration building and the support facilities building looking for the operators and possible fire. They retreated from the building when radiation levels up to 25r/hr (the highest readings on their meters) were detected. The reactor building was not entered.

Another entry was made after the health physicist (HP) arrived on the scene (approximately fifteen minutes after he was called). The HP and a fireman re-entered the administration building and support building and proceeded as far as the stairs to the reactor building operating floor. At this point, a radiation level of 25r/hr (meters limit) was detected and they retreated. They thoroughly searched the administration support buildings and found no fire, smoke or operators; however, they found radiation areas with levels ranging from 500 mr/hr up to 10 r/hr.

By the time the HP and fireman left the building it had been determined that the on-duty SL-1 operators had not left the site and it was presumed they were in the reactor building.

At approximately 9:30 p.m., two more HP's arrived at the scene with radiation survey instruments with a 500 r/hr range. One HP and another fireman went up the reactor building stairs until a 200 r/hr reading was obtained. They withdrew from

the building to plan action based on radiation levels. A quick entry to the top of the reactor building stairs was made. Radiation levels were as high as 500 r/hr here and a quick withdrawal was made. The team saw evidence of physical damage but no operators.

Key personnel were notified and had begun arriving on site. At approximately 10:30 p.m., entry was made to the reactor operating floor for two minutes during which time they saw two men, one moving.

More personnel re-entered the building and transported the man who was alive out on a stretcher to an awaiting truck. It was determined that the second man was dead. The truck carried the injured man to an ambulance where he was then transported to an awaiting doctor. The man was pronounced dead at 11:14 p.m. and the body was returned to the SL-1 site.

About 10:50 p.m., another group entered the reactor building and located the third man. He was lodged in the ceiling above the reactor and determined to be dead. It was later determined that the man lodged in the ceiling was in a radiation field of approximately 1000 r/hr.

The body of the second man was removed from the reactor operating floor on the afternoon of Jan. 4, 1961. Because of high radiation fields and the precarious position of the body lodged in the ceiling of the reactor building, actual recovery of the third body did not occur until January 9th.

The immediate cause of the SL-1 accident was determined to be the manual withdrawal of the central control rod of the reactor. The rod had previously been lifted and clamped in an up position. Every indication implies that at the time of the accident, there only remained the operation of removing the clamp and lowering the rod into its full-in position (in the core). From post-accident evidence, there is no apparent

reason why the rod should have been raised again. It is possible that it was raised because of potential sticking problems. The operator might have decided to just exercise the rod without any forethought to the consequences, or it might have been done in anger or careless fun. These thoughts are pure speculation and the reason for the second withdrawal can never be known.

This action (second withdrawal) gave rise to a severe and rapid nuclear transient. Withdrawal of this central rod brought the reactor from a subcritical condition through critical to a very rapid supercritical condition. The power is believed to have increased on a period of 4 msec until the excursion was terminated by steam voids and core vaporization at a power level of 2×10^4 MW. At this time the plates in the center of the core had reached a vaporization temperature of 3767°F. The formation of the steam void aided in termination of the nuclear transient, but also created a high-pressure region.

This pressure, which developed in the reactor vessel spread out in all directions, finally resulting in an acceleration upwards of the entire mass of water above the core. It appears that this force bulged the vessel badly, distorted the upper flange, and caused the vessel to rise slightly over nine feet.

Evidence indicated that two operators were standing on the reactor vessel top at the moment of the accident. One of the operators was thrown to the side as the vessel rose. The other operator was impaled by a control rod plug and was carried to the roof of the reactor room where he remained suspended. The third man was a casualty because of radiation and flying debris. Two of the men died instantly as a direct or indirect result of blast damage and the third man died of a fatal head injury several hours after the accident.

Kinetic energy of the accident that resulted in the damage and the death of the three operators was less than 1% of the total energy released.

The Sl-1 accident involved several different types of errors. These included, but were not limited to:

Design errors.

Instrumentation errors.

Operator/supervisor errors.

There has been much discussion of the SL-1 accident, its causes and especially its lessons, which are too numerous to discuss in full, but include:

It should be impossible for a reactor to be made critical by the withdrawal of a single control rod; there should be an audible warning of a valid neutron signal for any personnel working in a dangerous area or at a potentially hazardous task.

There should be an operator at the reactor control console with nuclear instrumentation activated, and in communication with the working group during core change operations, such as fuel or control-rod movements that could involve large changes in reactivity.

Each reactor should have an emergency locker, remote from the reactor, but clearly accessible by emergency personnel, and containing necessary emergency equipment.

Up-to-date operating instructions, along with check lists should be available for all personnel to ensure the safe and reliable operation of a device as complicated as a reactor.

CHALK RIVER ACCIDENT

The NRU Reactor, situated at Chalk River, Canada, was moderated and cooled by heavy water. This engineering and research type reactor was designed to operate at 200 million watts (MW) thermal. The reactor core consisted of 200 fuel assemblies. Each assembly was made up of five flat metal bars of natural uranium. Each bar was clad with aluminum contained in a 2.5-inch diameter tube. Initial loading was made by slipping aluminum sheaths over the bars and then drawing them through a die to press them into contact. Of the original fuel rods, thirteen failures had occurred along with considerable contamination of the aluminum tank system (due to fission products) and uranium plating out on the aluminum. Because of this, the monitoring system designed to detect faulty fuel elements was essentially useless.

On May 23, 1958, the NRU Reactor was automatically shutdown after a week of steady-state operations. The automatic shutdown occurred from an excessive rate-of-rise signal. The shutdown was investigated but no reason could be found for its occurrence. The shift supervisor decided to restart the reactor. After four attempts at restart, all of which were terminated by causes other than one of the safety features, the fifth attempt succeeded. Everything operated normal for five minutes, then the reactor automatically shutdown, again by a rate-of-rise signal accompanied by other alarms.

One of these alarms indicated high radioactivity in the coolant circuit. A momentary indication (flashing light) showed three rods were full out, although they should not have been. It was discovered later that some of the other signals were due to a pressure transient, which resulted from a violent failure of a fuel rod. It was speculated that the pressure transient in the reactor vessel caused the heavy

water level to change and, in turn, generated a signal for rod withdrawal. This rod withdrawal then caused the transient.

The reason for the rate-of-rise transient was not found. It was determined that just prior to the last trip (which was caused by a rate-of-rise signal), the rate-of-rise power level was correct, but had exceeded the demand level. No definitive reason for the power exceeding the demand level has been found. It is postulated that a faulty selector switch could have caused the problem. For this to occur, certain contacts would have had to malfunction when the operator switched from the low-power mode of operation to the normal-power mode of operation. This transfer was from a mode where the rate-of-rise of power was at 3%/second to one where the rate of rise of power was 1 MW/second.

It is further postulated that a cladding defect (possibly existing or caused by the first transient) allowed water to seep into the faulty element during shutdown. Then, when the reactor was restarted, the water-filled element burst because the faulty selector switch was at 3%/second instead of 1MW/second. This faulty element may have been detected if the failed fuel system had not been contaminated.

After the second shutdown, the staff prepared to remove the ruptured element. Normal element removal is accomplished by positioning the fuel removal cask over the element; the cask is then sealed to an uncapped hole by an extension on the end of an extendable bellows. This extension allows reactor water to flow up the bottom of the cask. The normally empty chamber in the cask is then filled with water and a valve at the bottom of the cask is opened. This results in a heavy-water-filled passage from the cask into the reactor vessel. A guide tube is driven down from the upper assembly to the fuel assembly, which is then raised into the upper shield where it is allowed to decay for approximately 15

minutes. During this time it is cooled by reactor heavy water rising through the guide tube. After this cooling period, the assembly is drawn up into the fuel removal cask where cooling continues by an internal recirculating system. Another area in the cask, which normally contains a new element, is rotated into position and the new element lowered into the reactor through the guide tube. After loading the new element, the guide tube is raised to its normal position in the upper shield area and the valve closed. The bellows is then removed and the cask shielding skirt raised. The fuel removal cask is transferred to storage by a crane.

The first damaged fuel element was easily removed. In attempting to remove the second element, the guide tube failed to drive down. It was decided, therefore, to remove the guide tube mechanism plug with the fuel. When the fuel element was raised into the cask, it was discovered that the heavy water had drained from the cask; thus, no cooling was available for the element. To ensure that the element was not blocking the flow, it was raised above the bottom valve, but no water appeared and the element would not drop back into the lower section. Refilling of the cask with water was attempted, but failed.

An attempt was then made to transfer the cask to a nearby area where a water hose was available for cooling. On the first try, the drive motor was stopped by a safety interlock. (This interlock was to prevent travel of the cask with the bottom area extended.) The extension was raised and the cask moved. nother interlock opened the valve at the bottom of the cask where there was not enough heavy water. Valve closure attempts failed because of yet another interlock which prevented valve operation unless the extension was down (it was up at this time). In other words, the interlocks were

so arranged that either position of the extension would cause trouble during this emergency operation.

During the time the aforementioned problems were occurring, the fuel element was becoming very hot. It had begun to fall apart. A small section of the fuel fell on the reactor top and a section approximately 3 feet in length fell into a maintenance pit. Both pieces began to burn. Finally, cooling water was connected to the fuel cask and it was transferred to a storage area. Due to the open valve, contaminated water leaked on the areas over which the cask transfer was made. Sand was brought into the facility and used to extinguish the burning uranium in the maintenance pit. It was fortunate that the larger section of the uranium fell into the pit, because of the shielding provided by the pit walls from the high gamma level radiation. A radiation meter held over the edge of the pit pegged at the top of its scale (1000 r/hr). It was estimated that radiation levels were as high as 10,000 r/hr.

Personnel exposure was limited, the highest being 19 r to one man. Very little radioactivity escaped from the building; however, the inside of the building was badly contaminated. The source of contamination resulted from fission products from the uranium fire. It took approximately 600 men two months to cleanup the building. Fuel fragments were dispersed in many areas, from the reactor tank to the fuel cask, on the reactor top and in the maintenance pit.

The heavy contamination as well as the limited exposure to personnel was the result of several factors:

Transfer of the hot element to a position less safe than the reactor tank (because of cooling).

Failure to follow policy of checking all equipment prior to usage to ensure its proper operation.

Removal of highly irradiated fuel - a very critical

safety problem. (Consideration for proper cooling of the element and actions to be taken in the event of a loss of coolant accident should have been evaluated. By transferring the fuel, two very effective containment boundaries, fuel cladding and primary containment, were negated simultaneously.)

Reactor design engineers should pay strict attention to functional aspects and modes of operation (in which safety interlocks are called upon to actuate) to ensure they do not interfere or cause problems when called upon to operate in other modes of operations.

THE WINSCALE ACCIDENT

The Windscale No. 1 Pile, situated in the United Kingdom, was an air-cooled, graphite-moderated, plutonium production reactor using natural uranium fuel slugs. The graphite, when bombarded with fast neutrons (neutrons with an energy > 0.1 MeV), undergoes extensive radiation damage. As a result of neutron bombardment of the graphite in the temperature range of 390° to 575°F, the graphite "grows" in size, its electrical and thermal conductivity goes down and it tends to store energy. This storage of energy by graphite, first suggested by Dr. Eugene Wigner, is often referred to as "Wigner Energy." After initial storage, if heated slowly, the energy is released as additional heat. That is, addition of some heat tends to trigger the release of more heat.

This unstable situation can cause or increase a safety problem in a graphite-moderated reactor operating below 575°F. Therefore, it is common practice to anneal the graphite. In this way, the pile is gradually heated in one manner or another, stored energy is gradually released and radiation damage,

for the most part, is annealed out. Release of this extra heat further increases the overall temperature of the pile. (This release, suggested by Dr. Leo Sziliard, is commonly referred to as "Sziliard Fever..")

At the time of the Windscale accident, two methods of graphite annealing were being used. In the United Kingdom, a "static" technique was used. Cooling air was shutoff and the temperature of the pile was increased by nuclear heating for a short time. Then the nuclear aspect of the pile was completely shut down. If cooling was needed, small amounts of air were admitted.

The Windscale accident began on October 7, 1957. The pile had been shutdown and the main blowers shut off in preparation for a Wigner release. All checks were completed to verify the pile was completely shutdown. The temperature devices (thermocouples) were checked and those inoperative were replaced. Nuclear heating began the evening of the 7th and was stopped early the next morning. Several hours later, it appeared to the operators that, instead of the temperature increasing as it should be, it was decreasing. It was decided to apply more nuclear heating. The second nuclear heating began on the 8th of October. The uranium thermocouples showed a temperature increase. Rods were inserted to reduce pile power and to allow the uranium to cool. On the 9th of October, the graphite temperatures showed considerable variations. One graphite temperature continued to rise steadily. At 10:00 p.m., the physicist in charge, because of the temperature increas, too certain actions. These actions included admitting cooling air to the pile for a short time. This cooling affected all the graphite temperatures except the highest one, where the rise was only slowed.

Early in the morning, on October 10th, the exhaust stack activity meter near the filter at the top of the stack showed a sharp increase. The physicist assumed this to be a result of the cooling air through the pile. Stack readings began to decrease for awhile, but showed increasing readings thereafter. Later that morning, graphite temperatures started to rise again. Cooling was again applied, with the stack meter showing sharp increases.

These indications suggested the existence of one or more burst cartridges. A decision was made to scan the pile for burst cartridges. It was then found that the scanning gear was jammed. Another method showed that there appeared to be a bad burst. A plug was removed in the pile wall giving a view of 4 uranium channels, including the one where the temperature reading was highest. The uranium cartridges were seen to be red with heat.

The staff first tried to remove these cartridges without success, and then tried to remove the surrounding cartridges. This succeeded in only containing the fire for some hours, but did nothing to reduce the temperature in the affected area.

During the night, CO_2 was used in an attempt to decrease the temperature of the affected area, but the region was too hot.

On Friday, October 12th, water was finally applied to the pile. The application of water continued for over 24 hours until, by Saturday afternoon, the pile was cooled.

As a result of this accident, the pile was a total loss. There was also some activity released to the surroundings, and considerable adverse publicity resulted because some iodine got into the British milk supply.

The Windscale accident resulted for many reasons:

Lack of instrumentation at the proper points for Wigner release.

Lack of operating staff knowledge of the weaknesses of their instrumentation; that is, a true picture of the situation was not provided from their instrumentation.

Lack of proper procedures to cover all operations (and actions of supervisors and operators working with inadequate information).

Inadequate design. (The effects of operation, on equipment, should be considered and equipment designed so that operation under all conditions is feasible. (Also, a system design with a positive feedback, that can cause or increase the possibility of an accident, should be avoided.)

ENRICO FERMI ACCIDENT

The Erico Fermi Reactor at Monroe, Michigan, approximately 30 miles south of Detroit, was a sodium-cooled fast reactor. The initial core was designed for 200 MW. The fuel was 10% (by weight) molybedenum alloy, enriched in ^{235}U to 25.6% (by weight). The fuel pins, clad with zirconium, were about 31 inches long and 0.158 inches in diameter. Proper pin spacing was obtained by placing 140 pins inside a square stainless steel tube with an internal grid structure. Each stainless filled tube was referred to as a subassembly. Each subassembly was then placed in the vessel where it was fixed radially at the bottom by a structure plate. The subassemblies were fixed vertically and radially at the top by a hold-down device. Spacer pads located about core mid-plane were

held in place by action of the hold-down device. The subassemblies were cooled by the liquid sodium flowing up from the bottom.

The sodium was directed through the subassemblies by a flow guide below the bottom of the structure plate.

The Fermi reactor achieved initial criticality on August 23, 1963, with a loading of 99 subassemblies. The High Power License for Fermi was obtained on December 17, 1965, and power level exceeded 1 MWt on December 29 of that same year. Fermi reached 100 MWt the following July.

In the fall of 1966 (October 4 and 5), a reactor operation throughout the night was conducted to heat the sodium to approximately 1 MWt. In the afternoon, October 5, a power increase began. At approximately 5 MWt a feed pump malfunction was noted and power was reduced to 2 MW. An auxiliary pump was put into operation and the power increase resumed.

At approximately 34 MWt, several abnormalities were noted; the control rods were withdrawn farther than necessary to obtain this power level and two core subassembly outlet temperature readings were higher than normal. Shortly after these abnormalities were noted, radiation alarms were received from the detectors located on the ventilation exhaust outlets from the reactor building. This situation initiated an automatic isolation of the reactor building and a radiation emergency was announced. The reactor was manually scrammed approximately 15 minutes after the first abnormality was noted.

The accident was caused by the breaking of a zirconium plate on the conical flow guide. This plate migrated to the bottom of the reactor where it came to rest over the sodium inlet holes in the bottom of the reactor vessel. Two of the inlet holes were completely covered and two other inlet holes were partially blocked. This blockage of sodium to the sub-

assemblies resulted in the heatup of four subassemblies.

Heating of the subassemblies resulted in massive melting of fuel pins in two of the subassemblies, with a general displacement of fuel in the axial direction. The center of mass was also shifted downward. The two subassemblies were fused, with one having physically ruptured.

Another subassembly showed that all fuel above the midplane exhibited phase changes, indicating that the temperature was in excess of 1100°F; however, no fuel-pin ruptures occurred. The fourth subassembly, in which flow was blocked (partially), showed that the fuel exhibited no phase change, but the subassembly was badly warped (bowed).

The accident was the result of a failure of a flow guide device installed to aid in reducing formation of a second critical mass in the event of a meltdown. The failure of this guide and its possible migration to the cooling inlet area should have been a design consideration. The fact that, at the time of the accident, the rods were out further than normally required should have been an indication of problems, and procedures should have been in effect to detail operator action if this occurred. Since the accident, the Fermi reactor and components have been dismantled and sent to various areas for burial or use in other facilities.

CRITICALITY ACCIDENTS

The accidents described above are not the only kinds of accidents that have occurred, nor are they the only ones. Other accidents or incidents have happened at similar facilities, both in the United States and abroad. An example of

this is the Water Boiler Criticality Excursion.

The water boiler, a 12-inch diameter stainless steel sphere, contained approximately 13.6 liters of a water solution of uranyl nitrate reflected by graphite. When the excursion occurred, the reactor was being remodeled for higher power operation. Two new control rods were placed in the system along with the three existing ones.

During normal operation, the control rods are raised remotely from a control panel (turned on by a key switch) located in the control room. At the time of the excursion, the control panel was deactivated and the individual responsible for building the new control rod mechanisms opted to test the drop times of the new rods. He opened the reactor top and manually lifted one of the rods, neglecting the possibility that this withdrawal could cause criticality. He tested each rod individually several times without a criticality occuring (one rod provided enough negative reactivity to maintain the reactor in a shut-down condition). The operator then decided to test the relative drop times of the two new rods and withdrew them simultaneously. They were held out for about 5 seconds and then released. The operator then repeated the test.

This second action caused the reactor to go prompt critical (critical on prompt neutrons only). The excursion was not immediately detectable because all control room instrumentation was de-energized except for one direct-reading thermometer. The strong negative temperature coefficient of the reactor is believed to have limited the transient prior to the control rods being reinserted. The operator received a gamma dose of about 2.5 r. There was no damage to the reactor.

Because this type of occurrence is possible, reactors should be designed so that it is impossible or extremely difficult to raise control rods by hand. Another restriction

should be that no employee be allowed to work alone or in any isolated operation that is potentially dangerous. Also, in any conceivable action that can result in a reactivity addition, proper instrumentation should be on and monitored by a qualified individual. There should also be communication between the monitor and the person performing the operation.

Another example of a criticality excursion occurred at the Alize' I Reactor in France. The Alize' I reactor, a light-water, pool-type reactor, was designed especially for criticality experiments at low power levels (1 to 100 watts). Different combinations of fuel types and fuel enrichment processes were available. Four control rods normally control power, with their configuration dependant upon the experiment in progress.

An experiment conducted in March of 1960 required that a stable period be established at a very low power level. Thus, criticality was established and the reactivity necessary to establish the required period was determined. This determination was made from the rod configuration and rod worth curves (curves showing reactivity worths versus rod positions). The control rods were then withdrawn to establish the required stable period. For some undetermined reason, a partially withdrawn rod was withdrawn to its full out position. This placed the Alize' I reactor on a very short period. The experimentor, who was working alone, may have withdrawn this rod full out due to a miscalculation of the reactivity addition or due to a number of other reasons.

After the rod withdrawal, reactor power rose on a period of approximately 1/2 second to almost 10 MW. Power was limited to this amount because of the Doppler effect of the fuel. This reactivity effect (Doppler) turned reactor power and caused it to decrease to just under 1 MW. Instrumentation

for the reactor was ineffective. Period protection trip circuits were bypassed, by mistake, and the power level trip did not function (possibly the instrument response time was too slow compared with the period). Actually, the first indication of a problem was from an adjacent reactor, which generated an alarm. Investigation of the alarm led the adjacent reactor operators to the Alize' I reactor as the cause of the alarm. Slightly over a minute after the peak of the power burst, the Alize' reactor was scrammed.

Withdrawing control rods to predetermined position is a normal activity at some reactors; however, actual period verification is carried out, as is monitoring of all instrumentation. Safety instrumentation and particular scram trips, necessary and essential to the operation of a reactor, should not be bypassed.

Various incidents have occurred at the Hanford Site in Washington State. At Hanford, a large group of graphite-moderated, water-cooled, plutonium producing reactors are available. Most information about the reactors and their operation is classified; however, the following incident has been declassified.

Before reactor start-up on October 3, 1954, a water leak had wet a large portion of the graphite in one of the reactors. Previous startups had been made with this condition and no problems had been encountered nor were any expected this time.

Startup was commenced and criticality was obtained. After an hour, the reactor was on a stable 120-second period and everything was normal. Calculations were made and control rods withdrawn to place the reactor on a 30-second period. The operator made a mistake, and the resultant reactor period was approximately 12 seconds instead of the required 30 seconds. The operator noted the mistake and began inserting

control rods. At approximately the same time, coolant outlet temperatures were noted to be approaching their limits; also coolant water pressure, of a large number of fuel tubes, was decreasing. The supervisor was informed of the situation and ordered more rods inserted. Shortly thereafter the reactor scrammed on low coolant pressure in the tubes. Since this type of scram had occurred before, reactor startup was recommenced in about an hour and the experiment proceeded without a hitch. During operation the following day, several fuel element failures occurred in the same region as the trouble the night before. Operation of the reactor continued until the fuel was spent.

Afterwards, it was found that two startup chambers were in close proximity to one another and in the same region as the water leak. This leaking water was assumed to have reduced the signal to both detectors. It was further assumed that leakage and location of detectors were contributing causes of the accident.

The supervisor, although experienced, was new at this particular job location and was not overly familiar with this particular reactor. At the time of the accident, the supervisor also was overseeing the shifting of scales of certain level instruments.

GENERIC DESIGN PROBLEMS

During nuclear power plant operation, various problems have been encountered more than once in the same type of operating reactor plant. These particular problems or occurrences are referred to as "generic" for that type of

plant.

The fuel in all current generation light-water reactors (LWR's) is uranium oxide (UO_2) in the form of small cylindrical pellets. The uranium is slightly enriched in the fissionable isotope, ^{235}U, and typically contains 1 to 4 percent ^{235}U with the balance being ^{238}U. In the manufacturing process, uranium oxide powder is compacted into pellet form and sintered at high temperatures to form a ceramic-like solid. This process yields pellets with densities that are less than the maximum possible density of void-free UO_2 (called the "theoretical" density). Typical pellet density for a PWR is approximately 90% of theoretical density and that of a BWR approximately 95%. The presence of some porosity in the pellets allows for swelling of the pellets as fission products (gases) accumulate in the oxide matrix.

Fuel pellets are stacked end-to-end in thin-walled Zircaloy tubes. (The Zircaloy tube is called the fuel cladding.) A standard fuel rod is about 13-14 feet long and contains a 12-foot column of stacked oxide pellets. Pellet diameters are such that they do not contact the cladding wall. The fuel rods stand vertically in the reactor, with the pellets inside the rods held down by springs. Space inside the fuel rods not occupied by fuel pellets is usually filled with a gas.

During operation, at both PWR's and BWR's, it has been noted that an increase in fuel oxide density has occurred. This "densification," which is a result of micro-structural changes in the oxide at very high temperatures, also has resulted in a decrease in pellet length and diameter.

In some cases, this decrease caused a gap between pellets and/or cocked pellets inside the fuel rod (Figure 7-3-1).

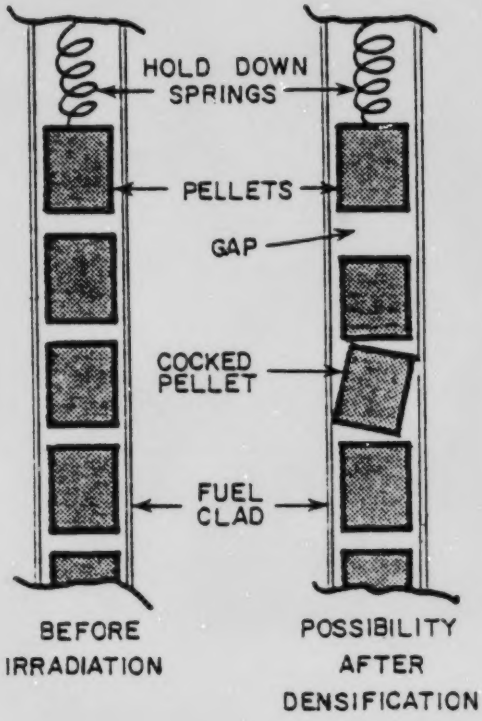

Figure 7-3-1. Before and after fuel densification.

Densification can cause problems, including:

Local power spikes, resulting from axial fuel column gaps.

Increased linear heat generation rates, due to axial shrinkage of the pellet.

Cladding collapse in area near gaps.

Increased stored energy, due to decreased pellet-cladding thermal conductance resulting from increased radial gap size.

It is expected that developments in fuel manufacture, such as in greater initial density and more complete sintering, can reduce the impact of densification. At present, operational limits on linear heat generation, design

evaluations on core related systems and, in some cases, power limitations are placed on those plants where densification has occurred with subsequent cladding collapse.

Many PWR's have been plagued with steam generator tube leaks. These leaks, which occur in the steam system and do not involve the reactor core itself, can result in leakage from the primary coolant system to the secondary system.

One solution, plugging the leaking tubes, which prevents water flow through them, has been accomplished at several operating nuclear plants with marginal success. Another solution, which has enjoyed limited success, calls for a change of the chemistry used to control corrosion.

If the problem continues, or the severity of the leaks increases, shutdown of the plants and replacement of the entire tube bundle becomes necessary.

Both BWR's and PWR's have experienced generic problems with their main steam relief valves. The pressure relief valves on a BWR are typically operated on a high-pressure signal, as sensed by a vessel-pressure sensor. This sensor applies a signal to a solenoid-operated pilot valve, which opens to permit air to open the relief valve. Ambient temperatures have allowed these valves to fail to operate or fail to close when required.

Pressurized water reactors are not without their relief valve problems. During hot functional testing at a South Carolina PWR, a relief valve was blown out of the steam line in which it was welded. This accident was attributed to a generic defect in the welds.

The failure of a valve to open could result in overpressurization of a reactor pressure vessel (in the case of BWR's) or overpressurization of steam lines (PWR's). If a valve failed to reseat, an uncontrolled cooldown condition could result, leading to excessive plant cooldown rates and

violation of the Technical Specifications.

Another generic problem shared by nuclear plants operating with seawater as their circulating water supply is the clogging and plugging of screens, intakes, condenser tubes and piping by shellfish, small fish and general marine life.

In an incident noted by a Southern Connecticutt BWR, during a routine startup, seawater entered the condenser system through many leaks in the main condenser tubing. At the time of the startup, 400 (of the 40,000) tubes in the condenser were plugged. Other tube leakage before this time was minor and high reactor water quality was maintained by 1) addition of "stopleak" to the condenser seawater intake and 2) by the high deionizing capacity of the demineralizer beds. But the magnitude of the leaks occurring during the startup, and the high ionic concentration of the incoming seawater, caused the demineralizer beds to rapidly become depleted, allowing seawater to enter the primary system. Chloride content of the water in the primary system exceeded the operational limit of 1 ppm (reaching a maximum of 15 ppm) with the primary system at rated operating temperature and pressure.

To prevent reoccurrence of this incident, the main condenser was retubed.

Cracks have been discovered in various systems in several operating BWR's:

Reactor water cleanup systems.

High-pressure coolant injection systems.

Reactor recirculation loops.

Core spray systems.

These cracks, detected by leak detection systems or through routine inspection and surveillance programs, have occurred in stainless steel piping.

OPERATING INCIDENTS

Incidents at operating nuclear plants run the scope from the mundane to the point where the plants' Technical Specifications are violated and/or forced outages occur. For example, an occurrence may be that, during routine surveillance, a pressure switch failed to actuate at the required setpoint because of instrument drift. Or, a procedural deficiency may have resulted in an accident or a reactor scram. Similarly, a component may fail, an oversight may occur, an operator may err. Most incidents or occurrences at nuclear power plants are no different from those at fossil-fueled power plants. But, controls and accountability of a nuclear plant places their incidents under public scrutiny (while those at fossil plants are seldom mentioned).

Even so, not all incidents which occur at nuclear power plants can happen at fossil plants; e.g., the release of radioactive gas from an operating PWR when venting a mixed-bed demineralizer.

The winter of 1977 provided some unique operating incidents at various plants throughout the nation. A number of plants had problems with sample lines freezing. A northern New Jersey plant had an ice plug form in a stack gas radiation monitor sample line because the heat tracing on the line failed. This item, insignificant as it may sound, caused the plant to initiate a plant shutdown; however, plant personnel were able to melt the plug (after dropping only about 200 MWe) and resolve the problem.

An Illinois plant discovered contaminated water overflowing onto the ground in front of the radwaste building basement. The source of water was found to be a resin-mixing tank inside the radwaste building. A 2-inch line from a contaminated demineralized water storage tank supplied the water

to the mixing tank. The absence of flow to the mixing tank indicated that a newly replaced stop valve was fully closed; however, an iceplug was located somewhere in the line. When it melted, the following day, contaminated water from the storage tank flowed into the resin-mixing tank, which eventually overflowed. Action was taken to reduce the possibility of reoccurrence.

A northern Illinois plant, during an abnormal stretch of cold weather, was operating in what is termed the ice melt mode. Even so, a large quantity of ice entered the forebay and fouled the traveling screens, which tripped the screens and two of the circulating water pumps. This reduction resulted in a violation of the Environmental Technical Specification temperature discharge limit and plant power had to be reduced by about 10 percent.

PLANT AVAILABILITY

As noted by the Atomic Industrial Forum (AIF), "nuclear power plants produced 12% of the United States electricity during the first six months of 1977; and at the end of June 1977, nuclear plants represented about 8.6% of the total installed generating capacity of electric utility plants in the United States."

The reliability of nuclear plants, on the average, is as good as, or better than, coal plants of the same age and size (Table 7-3-1). A few plants, generally early plants of a particular design, have had operating problems that have delayed startups or reduced operating times. In general, these problems are with conventional steam plant equipment and

TABLE 7-3-1

CAPACITY FACTORS AND AVAILABILITIES
FOR LARGE POWER PLANTS

(averaged over the years 1972 through 1975, except for plants started up during that time)

	Avg. Availability[1]	Avg. Capacity Factor[2]
Five Coal Units	70%	53%
Six Nuclear Units	69%	55%

[1]Availability = fraction of total number of hours per month that plant is available for service.

[2]Capacity Factor = amount of energy generated compared with amount that could have been produced if operated all the time at 100%

unrelated to their nuclear reactors. Nuclear plants in general, are on-line (available for operation) as steadily as other steam electric plants of comparable sizes (Table 7-3-2).

An up-to-date availability analysis by the authors of Power Engineering, presents availability as a function of load factor (load factor is defined as the ratio of energy actually produced, to the product of rated power times the reference time interval). This study shows that the load factor for nuclear plants in the United States increased from 60% in 1976 to 67% by the middle of 1977.

TABLE 7-3-2

AVERAGE PLANT AVAILABILITIES, 1964 THROUGH 1973

Type of Unit	Number of Units	Percent forced Outage	Percent planned Outage	Percent maintenance Outage	Percent Avail-ability
FOSSIL:					
200 to 289 MWe	234	4.4	6.5	3.1	86
390 to 599 MWe	99	7.7	8.7	3.8	78.8
600 MWe	59	14.0	8.5	4.5	73
NUCLEAR:					
130 to 1200	20	6.9	9.3	4.9	78.9

EXERCISES

Specific reactor accidents, other criticality accidents, and reactor incidents may be assigned to different students as research projects with both written and oral reports.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

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1. Describe at least three lessons learned from the SL-1 Incident (as discussed in this module):

2. Describe similarities between SL-1 accident and Water Boiler Excursion (as discussed in this module):

3. Define:

   Wigner Energy.

   Sziliard Fever.

   Densification.

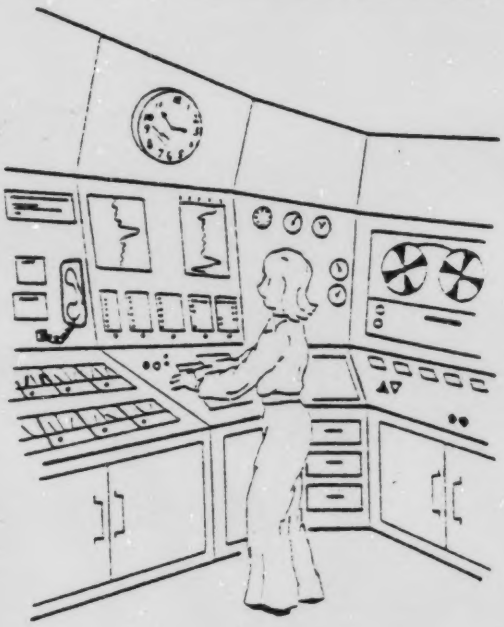
   Load factor.

4. List at least two generic problems common to BWR's and PWR's:

5. List three incidents at nuclear power plants attributible to the winter of 1976-77:

NUCLEAR TECHNOLOGY

MODULE 7-4 REACTOR STARTUP PROCEDURES



REACTOR
OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER – SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 7 REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

Power reactor startup, though highly complex, demanding and totally interrelated to the balance of the plant, is mostly a matter of appropriate procedures and properly sequenced (step-by-step) performance by reactor operator personnel.

Initial startup begins with "pretesting" of systems as much as a year or more before the reactor is actually brought up to power. Restartup, on the other hand, follows a much simpler format, and may be accomplished in a comparatively short time. (Some small research reactors experience startup-to-power in a matter of minutes. For a power reactor this process may require hours or even days.)

To exemplify the total startup process for a power reactor, this module describes a complete preoperational and startup test program for a boiling water reactor (BWR).

MODULE PREREQUISITES

None.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Outline a BWR startup program, in brief form.
2. Define:
   - Preoperational testing.
   - Fuel-loading procedures.
   - Heatup and low-power testing.
   - Power operation test program.
3. List four major preoperational tests.
4. List eighteen other preoperational tests.
5. Define:
   - Ascent to rated power.
   - Small perturbation checks.
   - Large transient checks.

Throughout most of a student's involvement with nuclear reactor study materials, the two major light-water reactor (LWR) types used in the United States to energize electric power generating plants are compared and contrasted. But in this discussion, a boiling water reactor (BWR) startup is described to the total exclusion of its counterpart, the pressurized water reactor (PWR). This exclusion is justified on two counts: 1) the essential procedures of a BWR startup are inclusive and sufficiently similar to PWR startup procedures; 2) it is not the details of a startup that a student should be concerned with, but the overview. (Each operator's or technician's specific tasks in a startup are specified in detail by each utility's operations manual, which incorporates procedures stipulated by equipment manufacturers and by various regulatory agencies.)

One major reactor manufacturer (General Electric) provides clear and specific guidelines for a BWR "Preoperational and Startup Test Program." Without implying any sort of endorsement, this program is used as the basis for this discussion. Major objectives of that program include:

Verification that all equipment and systems meet design and operational requirements (prior to fuel loading).

Demonstration of the nuclear steam supply system (NSSS) and the emergency core cooling system. (These systems must meet specifications through the full range of operating temperatures, pressures and power.)

Demonstration of reactor's ability to perform acceptably through all operating modes.

Demonstration of the ability of all startup and operational instrumentation and controls to perform according to Technical Specifications.

Allow operating (and technical) personnel to become familiar with reactor operating characteristics.

To verify performance of all safety systems.

To document test activities and results - this documentation becomes a part of the plant's permanent records.

Preoperational testing is performed by plant operating personnel. For initial testing, plant personnel normally receive assistance from a specially trained startup test crew provided by the reactor manufacturer and by technical representatives of various component manufacturers or designers.

Preoperational testing for the initial startup usually begins about a year prior to initial fuel loading. It ends with completion of a successful steam output demonstration - approximately 100 hours of continuous operation at rated conditions. Even before the preoperational tests begin, however, certain documents must exist:

Preoperational test approval (license) from NRC.

Preoperational test program specifications from equipment manufacturers.

Preoperational test instructions from manufacturers.

Startup test procedures and specifications from plant owner (utility company).

Startup test instructions from plant owner.

The purpose of each of these documents is rather self-evident. The content, however, may require some review. For

instance, the preoperational test program specifications provided by the reactor manufacturer define minimums necessary to ensure that the equipment meets those objectives listed previously. Also, they provide a brief description of each test, criteria for testing and a suggested sequence for performing the tests.

The instructions are "primarily working documents to be used by (the) operating and technical personnel" who actually conduct the tests. This documentation provides a step-by-step guide (based on extensive experience) for qualified operators and technicians to carry out the tests. It also includes necessary data logs for recording all pertinent information accumulated by the tests. Normally, equipment manufacturers provide data analysis methods and comparative data for quantifying and qualitating test results.

PREOPERATIONAL TESTS

Preoperational tests, as noted previously, are carried out on all plant systems. Most significant, however, are tests of the reactor and its energy (heat) transfer systems. Thus, the test program must verify that "initial fuel loading and subsequent operation can be safely undertaken."

As the tests progress, adjustments, modifications and even component and/or system exchanges may become necessary. All components are tested, logic checks are performed and sensor set points are determined or verified. Only when all its parts are working properly is the system tested as a unit. This preliminary test program includes:

Response and sensitivity measurements.

Wiring verification.
Air and hydraulic flow measurements.
Interlock checks.
Vibration measurements.

Only after this preliminary test program is completed is an integrated, functional test conducted. Simulated plant signals are imposed on the system to ascertain its response and intereaction; however, those systems that operate at elevated temperatures or in a high-pressure environment must be tested in their "cold" condition. (Final testing must await the system's initial nuclear heatup, which can be achieved only with an actual power ascension.)

Four major system preoperational tests are described below in some detail; other tests listed indicate the scope of such testing associated with a reactor - there are, perhaps, another hundred such tests associated with the steam cycle that must be performed independently.

The following descriptions are excerpted from General Electric's General Description of a Boiling Water Reactor, Section 10 "Preoperational and Startup Test Program."

REACTOR WATER CLEANUP

The vessel is cleaned and filled with high purity water and the reactor cleanup system is placed in operation to maintain water quality. In addition, all the process instrumentation required during the startup and "open vessel" testing is placed in operation.

NUCLEAR SYSTEM INSTRUMENTATION AND CONTROL

Each primary element, transmitter, receiver, indicator, control element, and any other instrumentation device is inspected. Any damaged or

inaccurate element is repaired or replaced. All interconnecting piping and wiring is verified. Each device is adjusted and calibrated in accordance with the manufacturer's recommendations. All control devices are exercised to ensure proper operation with the accuracy and response characteristics required by the system and specified by the manufacturer.

REACTOR PROTECTION SYSTEM CIRCUITS

Each circuit of the reactor protection system is tested to verify that a scram signal will initiate a reactor scram. For example, the reactor vessel pressure sensors in the reactor protection system are individually tested by varying a simulated pressure signal to the sensors. As a signal level corresponding to a scram condition is reached, scram functions and annunciation should appear for the particular channel under test. In this manner, all scram sensors except neutron flux sensors are tested.

Each intermediate and average power range monitor is tested to verify the scram function of these devices by an electronically generated signal. All reactor protection system coincidence features and the "fail-safe" feature of each component of this system are also tested for proper action.

CONTROL ROD DRIVE SYSTEM

The complete control rod drive hydraulic system (accumulators, scram valves, exhaust header, piping, pressure regulating valves, and interlocks) is tested without actuation of the drive mechanism.

The control rods are then attached to the drives, and rod guides are installed to test each rod drive and rod assembly individually for shim control and scram operation. The necessary instrumentation is provided to record the scram time for each rod drive. The operation is repeated as often as necessary to ensure that all the control rods are reliably operative.

OTHER PREOPERATIONAL TESTS

The following list, also excerpted from a General Electric publication, indicates by title other preoperational tests performed by operating personnel supported by manufacturer's teams, representatives and documentation.:

Feedwater control.
Standby liquid control (SBLC).
Nuclear boiler.
Residual heat removal (RHR).
Reactor core isolation cooling (RCIC).
Reactor recirculation system.
Fuel handling and vessel servicing equipment.
Low-pressure core spray system (LPCS).
High-pressure core spray (HPCS).
Off-gas system.
Fuel pool and containment pool cooling and cleanup.
Nuclear system leak detection.
Radwaste system.
TIP calibration system.
Rod pattern routine system.
Process radiation monitoring.
Process computer interface.

Containment isolation.

Preoperational testing ends when all items preceding fuel loading have been satisfactorily completed and all equipment is in good operating condition.

FUEL LOADING AND OPEN VESSEL TESTING

At this point, nuclear safety becomes a paramount concern rather than a consideration. In the following discussion, this concern dictates performance of every procedure. And every procedure is carefully defined in preoperational documentation.

The first step in fuel loading is to fill the reactor vessel with water to its specified level. Normal fuel-loading procedures are followed, using standard fuel-loading equipment. At this point (in an initial fuel loading) the fuel has not been irradiated; therefore, minor modifications may be made in the fuel-loading procedure, which is essentially identical for initial loading and reloading. It is not necessary, in an initial loading, to fill the fuel pool or the pool above the reactor. This omission allows the fuel to be handled more easily and directly.

Radiation sources must be placed in the vessel before fuel loading commences. Sb-Be sources, which produce neutrons (γ, n), and an Am-Be source are installed. The neutron sources, which are continually regenerated by rector operation, provide enough emissions to be read on the sensitive chamber used during fuel loading. The Am-Be source produces neutrons (α, n) in sufficient quantity for these chambers to read through all stages of fuel loading. (These special neutron-sensitive

chambers for initial fuel loading, installed around the periphery of the section to be loaded, can initiate a scram if reactor power rises above preset limits.

At every stage of fuel loading, the low-level nuclear instrumentation is tested for responses to nuclear flux. These scram sensors must be responsive:

High water level.
Manual scram.
High flux.
Low water level.

Throughout the loading process, the control-rod system must be operative, providing a "cold shutdown" margin. (Note: The standby liquid control system must be available during loading, critical testing and fully loaded condition.) As the core is loaded, step-by-step, adequate subcriticality tests are performed before and after each step. Such testing includes functional testing of each control rod and its drive. Simultaneously, the shutdown margin is measured and demonstrated to ensure that shutdown criteria are met.

When loading is complete, operational tests are begun.

HEATUP AND LOW-POWER TESTING

With the fuel in place, the reactor is brought critical for heatup and low-power testing. At this point, responses of startup range and intermediate range monitoring (instrumentations) systems are determined. If properly calibrated initially (from nuclear data), their responses should be appropriate. If not, calibration must be accomplished prior to closed vessel testing.

The vessel is closed and the nuclear system is hydrostactically pressurized and checked for leaks. Then, the entire plant is brought to a "condition of readiness for power operation."

The hydrostatic testing recommended by General Electric includes:

Operation of recirculation pumps.

Bringing of system to critical and beginning of system heating.

Calibration of intermediate and average power range monitoring systems, based on thermal data.

Pressurization of system.

Measurement of vessel surface temperatures.

Measurement of thermal expansion.

Testing of control-rod drive system (at various pressures).

Surveying of radiation.

Placing in operation the vacuum, condenser and off-gas systems.

Placing in operation cleanup, condensate, feedwater and other systems, as appropriate.

Operation of the reactor core isolation cooling and high-pressure core spray systems - it must start from cold condition and reach rated flow within specified time.

Rolling the turbine as soon as steam production is adequate.

At this point, reactor power is between 1% and 10% of full power.

POWER OPERATION TESTS

Full power is attained in incremental steps, with testing of all operational and safety features at every step. Normal increments are 25%, 65% and 100% of full power. A power operation test program is a vital part of the startup procedures prescribed by equipment manufacturers, utility operational and QA/QC test programs, and by regulatory agencies (e.g., NRC).

ASCENT TO RATED POWER

The following tests are major among the many performed during ascent to rated power:

- Calibration tests and adjustments of nuclear instrumentation (using heat balance techniques).
- Radiochemical samples and analyses (to establish base conditions, and check out laboratory procedures).
- Calibration of local power range monitor (LPRM) instrumentation system - beginning with flux measurements and resulting in correction factors for resetting the LPRM amplifier gains.
- Evaluations of core performance at each of the following: 1) significantly different power levels; 2) significantly different rod pattern; 3) significantly different flow rate; 4) significantly different subcooling rate. (The objectives: to determine 1) core power distribution, 2) maximum heat flux and 3) minimum critical power ratio at each different condition.)
- Operation - one at a time - of reactor relief valves to ensure that each meets specified capacity and operating characteristics.

SMALL PETURBATION CHECKS

These small peturbation checks are performed at each power increment. The objective is to ensure proper instrumentation and controls response to the following kinds of small disturbances:

Quick variation (± 10 psi) in pressure regulator set point.

Continuous control-rod movement (±) sufficient to vary local incore flux less than 10% of point.

Quick (±) changes (less than 10% of point) in recirculation flow.

Quick (±) changes (less than 10% of point) in feedwater flow.

LARGE TRANSIENT CHECKS

Large transient checks, because of their difficulty, are performed only at a few power levels. Their purpose is to demonstrate appropriate (safety) responses to abnormal disturbances:

Generator breaker trip; turbine control valves must close quickly and bypass valves must open instantly.

Turbine stop valve trip; bypass valves must open instantly.

Recirculation pumps trip; first one, then the other, then both.

Recirculation flow varied (at maximum rate over maximum range).

Main steam isolation valves closed.

Each of these tests causes a major transient, sufficient to activate one or more safety features, verifying proper response in the event of an unscheduled major transient.

NSSS PERFORMANCE TEST

After all startup tests are completed, the reactor manufacturer conducts a 100-low running test of the nuclear steam supply system (NSSS). This test demonstrates that the system produces steam at specified flow rates, at specified pressures and of the specified quality.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

General Electric, Nuclear Energy Division. General Description of a Boiling Water Reactor. San Jose, CA: General Electric Co., 1976.

1. Outline a BWR startup program; use headings from this module:

2. Define:

   Preoperational testing.

   Fuel-loading procedures.

   Heatup and low-power testing.

   Power operation test program.

3. List four major preoperational tests:

4. Define:

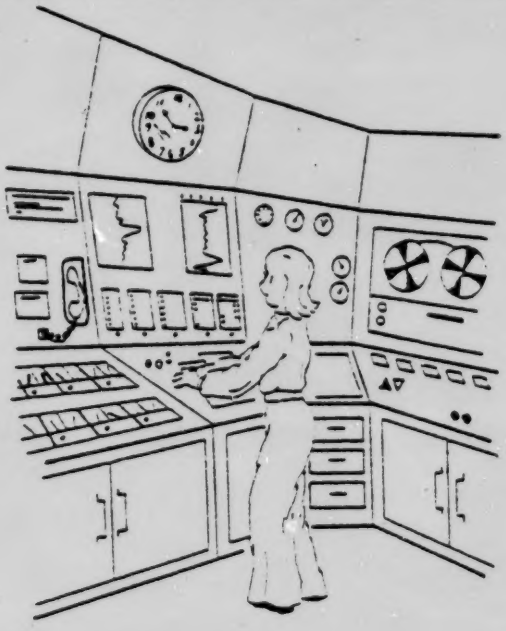
Ascent to rated power (list typical increments given in this module).

Small pertubation checks.

Large transient checks.

NUCLEAR TECHNOLOGY

MODULE 7-5 PRESSURIZED WATER REACTOR OPERATING CHARACTERISTICS



REACTOR OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER – SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 7 REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

As reactor plant technology developed in this country, two major types of power plants evolved. Military applications primarily were pressurized water reactors (PWR's) while commercial development included both PWR's and boiling water reactors (BWR's). In this module, operating characteristics of a commercial PWR are described.

MODULE PREREQUISITES

The student should have completed a course in reactor physics, Course 5, "Introduction to Nuclear Systems" and all previous modules in this Course.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List three major PWR plant configurations.
2. Describe methods used to control reactor power.
3. List thermal and hydraulic operating parameters of a PWR.

4. State general parameters controlled to limit reactor power.
5. Describe circumstances that cause xenon oscillations, including typical methods for compensation.
6. Enumerate major effects of temperature coefficients of reactivity (relative to core age, moderator temperature and fuel temperature).
7. List typical reactor trip parameters; give examples of typical values.
8. Describe effects of changing steam plant power on reactor system power.

REVIEW OF PLANT SYSTEMS

The primary function of a reactor coolant system (RCS) is to remove heat generated in the fuel by fission. Its secondary function is to transfer that heat through the steam generators to the secondary plant. The RCS acts as a carrier for soluble neutron poison (boric acid) and as a second barrier against fission release to the environment.

A major problem in the operation of a PWR is control of pressure in the primary system. If the pressure is too low, steam bubbles form at the point of lowest pressure, probably at the reactor coolant pump suction, and can cause damage to the pump. If the pressure is too high, a rupture in some primary system boundry might occur, which could cause a decrease in plant pressure. In either instance, a loss of reactor coolant flow occurs, with subsequent loss of ability to transfer heat out of the reactor core by normal means.

When pressure is below the desired value, a pressure-control system energizes pressurizer heaters. Electrical energy heats the water in the pressurizer until the desired pressure is attained. In most reactor plants the pressurizer heaters are divided into two or more groups, one for normal pressure control and others for more extreme plant pressure transients.

When plant pressure is not maintained in the "normal band" then more extreme measures are taken to protect the reactor core from damage. "Scramming" reduces heat generation from the core to a level that does not produce immediate danger. Table 7-5-1 is representative of the pressure control and protective setpoints used in a large PWR reactor plant.

TABLE 7-5-1
PWR SET POINTS (TYPICAL)

PRESSURE	CONTROL OR PROTECTIVE SETPOINT
2480 psig	Pressurizer Relief Valve Opens
2380 psig	High-Pressure Reactor Scram
2300 psig	Spray Flow - Full
2260 psig	Spray Flow - Starts
2250 psig	Group 1 Heaters - Off
2230 psig	Normal Pressure
2210 psig	Group 1 Heaters - On
2200 psig	Group 2 Heaters - Off
1980 psig	Group 2 Heaters - On
1970 psig	Low-Pressure Alarm
1800 psig	Low-Pressure Scram
1700 psig	Safety System Actuation

CHEMICAL AND VOLUME CONTROL

Functions of a chemical and volume control system (CVCS) include:

Adjustment of boric acid concentration in reactor coolant for reactivity control.

Maintenance of proper water inventory in reactor cool-

ant system.

Reduction of quantity of fission and corrosion products (maintenance of reactor coolant chemistry within design limits).

This system, which consists of a flow path for reactor coolant out of the primary system, diverts coolant to a water treatment system or pumps it back into the reactor coolant system. (Note: This system is called the makeup and purification system in some plants.)

SECONDARY SYSTEM

The primary function of the steam plant is to convert the thermal energy of steam into electrical energy out of the turbine generator in the most efficient, practical manner. A secondary function of this system is to act as a heat sink (to ensure that heat may be removed from the reactor core, and to limit the temperature of the reactor to a safe value).

TURBINE/GENERATOR

The turbine/generator receives steam into its pressure stages by way of "stop" valves and throttle valves. Steam exits its high-pressure section at a lower pressure; at this point, it contains moisture. This "wet" steam passes through a moisture separator (where most of the water is removed). Dry steam then passes into the reheater where it is reheated. The reheated steam (with a small amount of "superheat") provides more efficient operation of the low-pressure turbine sections.

CONDENSER

When "used" steam exits from the last stage of the low-pressure turbine, it enters the condenser. There it is condensed to water by transferring heat to the circulating water system. The waste heat is dissipated to a body of water such as a river, lake or ocean, or it is rejected to the air by cooling towers.

MAIN FEEDWATER PUMPS

Condensate pumps take a suction from the condenser hotwell and transfer water to the main feed pumps. The condensate water is usually directed through feedwater heaters where it picks up some heat from waste steam and hotter water.

Main feedwater pumps take suction from the discharge of the condensate system. The water, when discharged from the feed pumps, is called feedwater. Feedwater pumps (multistage, high-capacity, high-pressure pumps), can be electrically driven or driven by separate steam turbines. Their outlet pressures are controlled to be 100 to 200 psi higher than in the steam generators. After feedwater leaves these pumps it goes through high-pressure feedwater heaters, which serve the same function as previously explained.

SAFETY SYSTEMS

PWR safety systems are of two general types:

Low-pressure injection.
High-pressure injection.

Low-pressure injection is "passive," requiring no electrical power for actuation or for moving water into the core.

It is accomplished by a pressurized tank of borated water connected to reactor inlet piping, with a check valve to prevent flow until reactor coolant pressure decreases to less than that in the injection tank.

High-pressure injection consists of multi-stage injection pumps, redundant flow paths to reactor coolant system, and more than one source of borated water. Isolation valves in the flow paths are arranged in series-parallel. If a valve function is to initiate flow by opening, two parallel valves are used (with separate power supplies) and either valve can allow designed flow. If a valve function is to stop flow, two valves are arranged in series (with separate power supplies); when given a "close" command, at least one valve will close and stop the flow.

REDUNDANCE

Redundancy is often discussed in relationship to the operation of a nuclear power plant. In simple terms it means that all crucial components and functions are duplicated, either by other independent systems or by additional components of the same type with independent power supplies and control power. A statement often made is that no single component can cause or effect the failure of a critical safety function. Much of any Preliminary Safety Analysis Report (PSAR) or the Final Safety Report (FSR) concerns itself with plant design to meeting redundancy requirements. Each plant is tested frequently to be sure that all components affecting safety are operational. If not, the plant is required to shutdown until these requirements can be met without exceptions.

CONTAINMENT

The containment structure is designed to contain contents of the reactor coolant system in the event of a catastrophic failure of a large pipe in the reactor system. This building is designed to withstand a pressure higher than the maximum credible accident (MCA). The design pressure of most containment structures is about 60 to 65 psi. A containment building is constructed of concrete and reinforcing steel rod.

Other systems in the containment building include a spray system. If safety system logic detects an abnormally high pressure inside the structure, this spray system is activated. Water, containing sodium hydroxide, is sprayed into the containment atmosphere from spray line pipes at the top of the building. The sodium hydroxide combines with any iodine in the atmosphere (from reactor fuel), making it soluble, and isolating it in the sump.

Another system in the containment building, used in the event of an accident, is the containment cooling system. Its purpose is to cool the atmosphere and condense steam to reduce the pressure.

REACTOR CONTROL

A PWR is controlled by two methods of reactivity changes. The most readily available and the quickest acting is reactivity control by moving control rods. In most PWR's control rods are withdrawn from the top of the core. Pulling rods out implies upward motion.

Control rods are made of various materials, depending on

manufacturer (e.g., boron-glass, silver-indium-cadmium, and hafnium). All of these substances have thermal neutron absorption cross sections of from 10^3 to 10^5 barns.

A second method of reactivity control used in a PWR involves a chemical reactivity "shim." In this method, boron content in the reactor coolant system is changed. The property of boron used is the thermal neutron absorption cross section of $^{10}_{5}B$. Boric acid is dissolved in the reactor coolant. Its concentration is controlled to gain the desired reactivity.

When water is diverted out of the reactor coolant system it is "held-up" in tanks and then processed through boric acid evaporators to remove boron. Boric acid removed in this part of the cycle is stored in boric acid storage tanks for reuse.

REACTOR EFFECTS

Almost any change in operating conditions causes a change in neutron population (critical condition) of the reactor core. The major parameters of concern are moderator temperature and pressure, fuel temperature, and void (steam) formation. The effects of temperature are predominant, and they present the largest magnitude of reactivity.

MODERATOR TEMPERATURE COEFFICIENT

Moderator temperature coefficient is that reactivity change caused by 1°F (P/1°F). The primary cause of a reac-

tivity change is change in the amount of leakage neutrons as moderator temperature changes. For example, if by raising reactor coolant temperature 1°F, less dense water allows more neutrons to leak out of the core, a negative reactivity is seen. In this example, a positive change in temperature causes a negative change in reactivity - the typical response - which is called negative temperature coefficient (NTC).

A design criterion, imposed by law, is that a power reactor must always have NTC.

PRESSURE COEFFICIENT

Pressure in a reactor coolant system affects the critical condition of its fission reaction. If pressure is increased, more molecules of water are in the core area, which causes more neutrons to be thermalized. (Fewer leak out of the core; more cause fission.)

Magnitude of this potitive pressure coefficient is small, about 0.0001/100 psi. Potentially, it represents a dangerous situation. Increased power and reactor temperature can cause pressure to rise further; however, an effect is more than countered by the NTC (discussed previously).

POWER COEFFICIENT

Another effect of temperature is seen in the reactivity balance when fuel temperature is changed because of a change in reactor power level. When power is raised, the peak temperature in the fuel increases to provide increased heat transfer out of the fuel rod into the coolant. The main effect of this increase is to add negative reactivity. This effect primarily causes a change in the neutron absoprtion

cross section change of uranium-238 with temperature. U-238 has discrete absorption peaks at certain energy levels as neutrons undergo the process of thermalization. If the temperature of uranium fuel is increased, peaks broaden, and more of the energy spectrum of neutrons is susceptable to this non-fission producing neutron capture by uranium. Thus, as temperature is raised a net negative reactivity is seen.

The fuel temperature effect on reactivity is often called "doppler" (after the physical effect, as when a train approaches with whistle blowing, resulting in an increasing pitch or frequency.)

VOID COEFFICIENT

In a PWR, the ultimate safety margin is provided by formation of steam if the reactor overheats. If the temperature of the moderator reaches saturation for the pressure present, then the fission process is interrupted because of a lack of moderator. Steam has about 1/1000th the density of water so, if reactor temperature is being raised by a power excursion, formation of steam results in a very large mangitude NTC. This inherent safety margin is not considered part of the safety analysis of a PWR.

XENON

The xenon-135 isotope has a very large neutron cross section for absorption (2 million barns). Produced as a by-product of fission, it decays by beta emmission to ^{135}I, which does not present a significant neutron poison effect. After a reactor has been shutdown for 30 hours, xenon present at the time of shutdown is no longer present - it has

decayed by radioactive emmision - and the reactor is said to be "xenon free."

Operationally xenon causes two types of problems. Changing plant power level creates xenon transients, which have to be counteracted by changing the boron concentration. The other problem is encountered late in the core cycle when boron concentration is low. Shutdown of the plant produces a xenon peak, which adds more negative reactivity than can be compensated by removal of boron. Thus, with all control rods withdrawn and boron at 0 parts per million in the coolant, the reactor remains shut down (called "xenon precluded startup").

POWER DISTRIBUTION

A major thrust in power reactor operation is to obtain and maintain even power distribution in the reactor core at all times. The goal is to prevent the failure of fuel cladding. No matter how well a core is designed, if operated outside its designed limits, areas of excessive power generation (hot spots) can occur.

Operating conditions that contribute to uneven power generation include long-term plant operation with misaligned (or dropped) rods, which creates an uneven neutron flux profile in the core.

Core physics involved in this problem is complicated by xenon transients set up in the core when it is operated with different parts at different powers.

Recovery methods used to minimize (or eliminate) effects

of unbalanced power distribution include plant shutdown or power reduction to a point at which effects are not damaging to core. Either method requires decreasing plant output, which is undesirable.

PWR's normally are designed to be operated with all reactor coolant pumps running at 100% power. The automatic protection system ensures that this happens by preventing full power operation without full flow - the plant is tripped off if such a condition is detected or anticipated.

Power-to-flow ratio is equal to 1 at full power. Any change in this ratio indicates an unsafe condition, especially if flow is lowered while the plant remains at full power.

ACCIDENT ANALYSIS

An accident analysis is performed for all reactor plant licensing studies. The worst transients are assumed to happen when plant is operating as close to safety limits as procedures allow. In the case of a power transient, the worst case occurs when maximum power is reached during the transient. When this peak power is compared with the power needed to cause core damage, it is clear that the former must be less than the latter. This ratio, called design power ratio (DPR) is expressed as follows:

$$DPR = \frac{\text{Power reached during transient}}{\text{Power needed to cause core damage (minimum)}}.$$

Most plants limit the DPR to 1:3. As long as accident studies indicate a DPR greater than 1:3 the plant is safe

to operate.

High power is prevented by automatic scrams at about 25% above nominal during startup. Power is detected by at least four power-range nuclear instrumentation channels; usually two out of four of these channels exceeding the setpoint causes a trip. At higher power levels, the trip point is about 110% of nominal power, if plant is operated with full reactor coolant flow.

HIGH CORE TEMPERATURE

Excessively high temperature in the reactor core is detected by measuring delta t (reactor coolant outlet temperature minus reactor inlet temperature) and comparing it with allowable delta t relative to average temperature and pressure. (Normal delta t in a PWR operating at full power is about 50°F. Setpoints for delta t trip are about 65°F.)

PRESSURE

Pressure, in the primary plant, has one of the most immediate effects on plant operation. Protection against too-high or too-low pressure is straight forward. If the pressure approaches setpoint of primary relief valves and continues to increase, something is wrong and the reactor is scrammed. This scram occurs at approximately 100 psi below the valve setpoint, or 2380 psi.

When pressure drops below normal and the energizing of all heaters in the pressurizer fails to terminate the transient, the reactor is scrammed. This low-pressure setpoint is typically about 1800 psi.

SAFETY SYSTEM INITIATION

Indirect (preventive) safety systems are initiated when they sense conditions that preceed or predict a serious accident or unsafe condition.

Any unexpected loss of plant equipment, such as reactor coolant pumps, power to vital electrical busses or turbine generator, and loss of other necessary components can cause the plant to be tripped off prior to degradation of any plant operating parameters. Automatic actions of this type occur because the equipment lost is going to produce a trip later anyway, and no normal operator action can prevent it.

STEAM GENERATOR MALFUNCTION

The steam generator acts as the normal method of removing heat from the primary plant after the plant has been shutdown. If for some reason it is incapable of removing the heat, the primary plant must be cooled by some other means. The low-pressure shutdown cooling systems or, as a last resort, the safety systems can be used. Protection of the steam generator's ability to act as a heat sink is accomplished by tripping off the plant if water in the steam generator reaches a low value (minimum mass required to remove expected heat after shutdown).

TURBINE GENERATOR MALFUNCTION

The possibility of a malfunctioning turbine generator causing an accident receives much design and operational consideration. A problem in the turbine generator complex causes the generator breaker to open suddenly unloading the

turbine rapidly. The turbine then speeds up because steam is being supplied against no load. If unchecked, this speed results ultimately, in turbine blades being thrown off through the casing, causing damage to the reactor systems.

Design of the power plant physical floor plan takes this possibility into account. The protection system interacts to limit the availability of steam (in the system) to contribute to an overspeed excursion. In the sequence of events after a turbine generator problem, the reactor is scrammed and the main steam isolation valves are closed.

MAJOR STEAM RUPTURE

Major steam ruptures are possible in any type of power plant that uses steam as part of its cycle. Major hazards of steam system ruptures in a PWR are 1) the possibility of losing cooling capability for the primary plant and 2) damaging the containment building by releasing large amounts of steam into the building (causing it to be overpressurized and its integrity to be compromised).

The protection system senses steam flow and pressure to detect conditions that could cause a major break.

OPERATOR FUNCTIONS

The task of the operator is to monitor the safety systems for proper operations and to take corrective actions in the event of a malfunction. The operator has, as an option, a capability for initiating manual scrams, turbine trips and any

other safety system initiation if, in the operator's judgement, the reactor is in danger.

PWR MANUFACTURERS

In the United States, three manufacturers produce pressurized water reactors (Westinghouse, Combustion Engineering and Babcock and Wilcox). Concepts in this module apply to plant design of all three.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

None.

1. Briefly describe methods used to control reactor power:

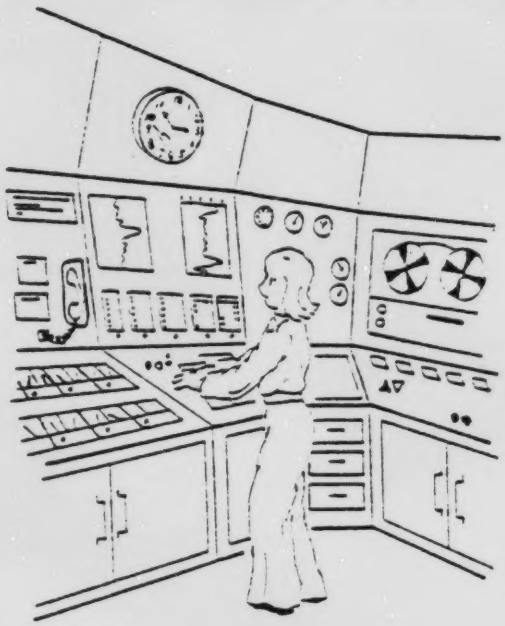
2. Briefly describe circumstances that cause xenon oscillations; include typical methods for compensation:

3. Briefly describe the effect of a steam plant power change on reactor system power:

NUCLEAR TECHNOLOGY

MODULE 7-6 BOILING WATER REACTOR OPERATING CHARACTERISTICS. -

RODNEY D. DAVIDSON



REACTOR
OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER - SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE 7 REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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INTRODUCTION

All thermal power plants, whether fossil or nuclear fueled, ultimately generate electricity in much the same manner. Energy in the form of steam is used to spin a turbine which drives an electrical generator to produce the power required by homes and factories. The major difference between the plants is the method used to generate the steam. Fossil fuel plants fueled by oil, coal and natural gas utilize the heat of combustion to convert water into steam. Nuclear plants, such as pressurized water reactors (PWR's) and boiling water reactors (BWR's), use the heat created by fissioning, or splitting apart, of uranium atoms to change water into steam.

This module describes how a BWR accomplishes its purpose, including major characteristics it exhibits.

MODULE PREREQUISITES

The student must be familiar with basic terms and concepts of reactor theory, heat transfer, fluid flow and thermohydraulics.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe the following as they relate to a boiling water reactor power plant:
   - Nuclear steam supply system (NSSS).
   - Nuclear boiler and internals.
   - Recirculation system.
   - Control-rod drive system.
   - Main steam system.
   - Turbine generator and control system.
   - Condensate and feedwater system.
   - Feedwater level-control system.
2. Describe processes involved in BWR power control.
3. List emergency core cooling systems.
4. Describe purpose of each BWR containment component listed below:
   - Fuel.
   - Reactor vessel.
   - Drywell and drywell floor.
   - Pressure-suppression chamber.

BOILING WATER REACTORS

A boiling water reactor system (Figure 7-6-1) is divided into two major areas:

Nuclear steam supply system (NSSS).

Balance of plant (BOP) systems.

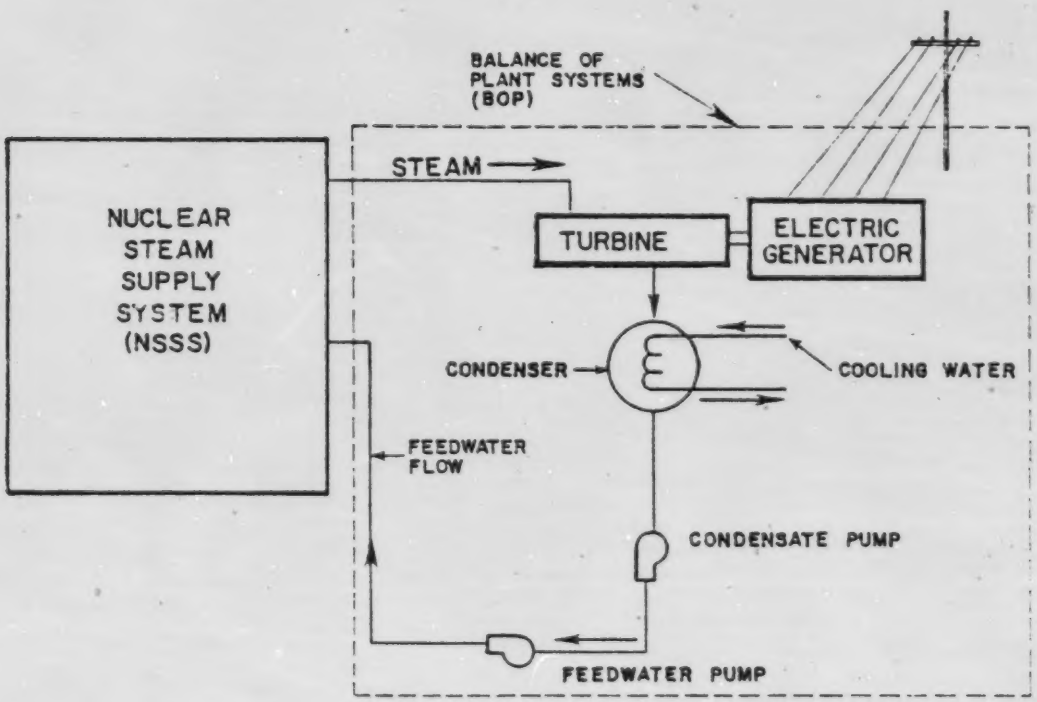

Figure 7-6-1. BWR.

The major purpose of the NSSS is to generate steam to spin the turbine. The purpose of the BOP systems is to accept the steam, use it to generate electricity, convert it back into water and pump it to the NSSS so that the cycle can be repeated.

NUCLEAR STEAM SUPPLY SYSTEM

In a BWR nuclear steam supply system (Figure 7-6-2), steam that has been condensed in the main condenser is pumped (as water) to the reactor vessel through a feedwater return

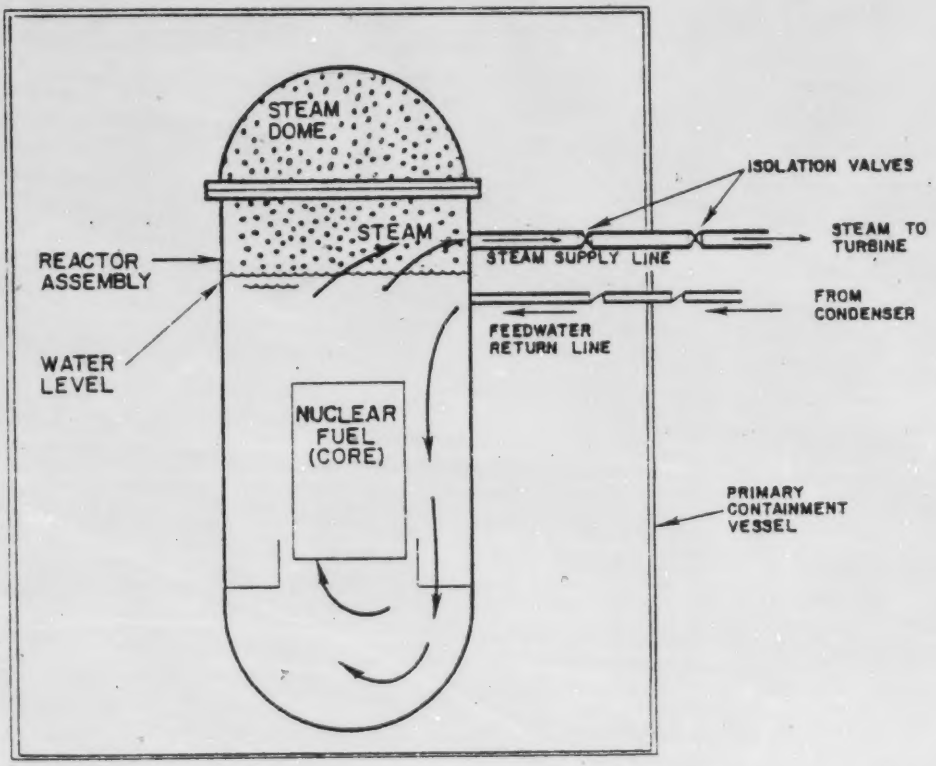

Figure 7-6-2. BWR nuclear steam supply system (NSSS).

line. In the reactor vessel, water is converted to steam by heat from the nuclear fuel, then sent to the turbine via the steam supply line.

Isolation valves, in both the steam and feedwater lines, limit the amount of radioactivity that can escape to the environment if either line breaks outside the primary containment. (The primary containment vessel limits the amount of radioactivity that can escape to the environment if a line breaks inside the containment structure.)

The "heart" of an NSSS is the reactor assembly (Figure 7-6-3). It consists of a reactor vessel, internal jet pumps,

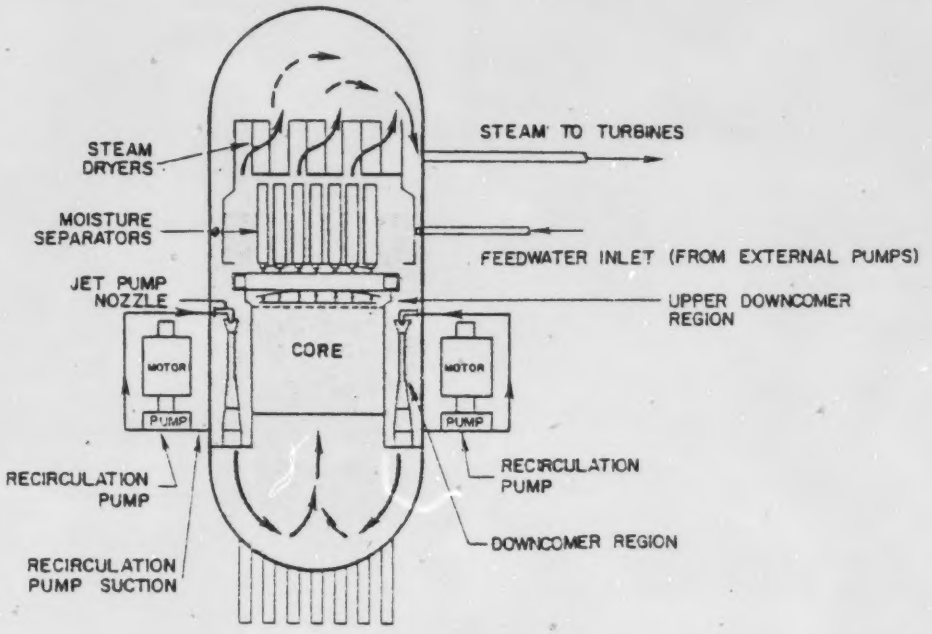

Figure 7-6-3. Steam and recirculation water flow paths through reactor assembly.

control rods and their drive mechanisms, core components and steam separator and dryer assemblies.

Water returning to the reactor (via feedwater lines) flows downward to the top, or suction, of the jet pumps. There, it is sucked into the pumps and forced into the region directly under the nuclear fuel. It then flows up into the fuel region where nuclear heat converts part of it into steam.

Steam-water mixture exits the fuel region and enters the steam separator assemblies, which - through a whirling action - remove most of the water and pass the steam on to the dryer assembly. The steam dryer forces the steam-water mixture on a zigzag path, which removes remaining moisture. Dried steam exits to the turbine via the steam supply line. (Moisture removal from steam is very important because water striking the fast-moving turbine blades causes rapid blade erosion.)

Moisture removed by the separator and dryer assemblies mixes with feedwater and returns to the jet pump suction (called "downcomer" region), where the cycle is repeated. A portion of the downcomer region water provides suction for the reactor water recirculation system (Figure 7-6-4). The recirculation system has two main functions. First, it forces additional cooling water through the core region, which increases ability to remove heat generated by nuclear fuel, resulting in more power being generated by core. Recirculation pumps accomplish this by taking a suction from the downcomer flow in the annulus between core shroud and reactor vessel wall. Approximately one-third of core flow is taken from the vessel through two recirculation lines, pumped to a higher pressure, distributed in a manifold to which a number of riser pipes are connected and returned to the recirculation inlet nozzles.

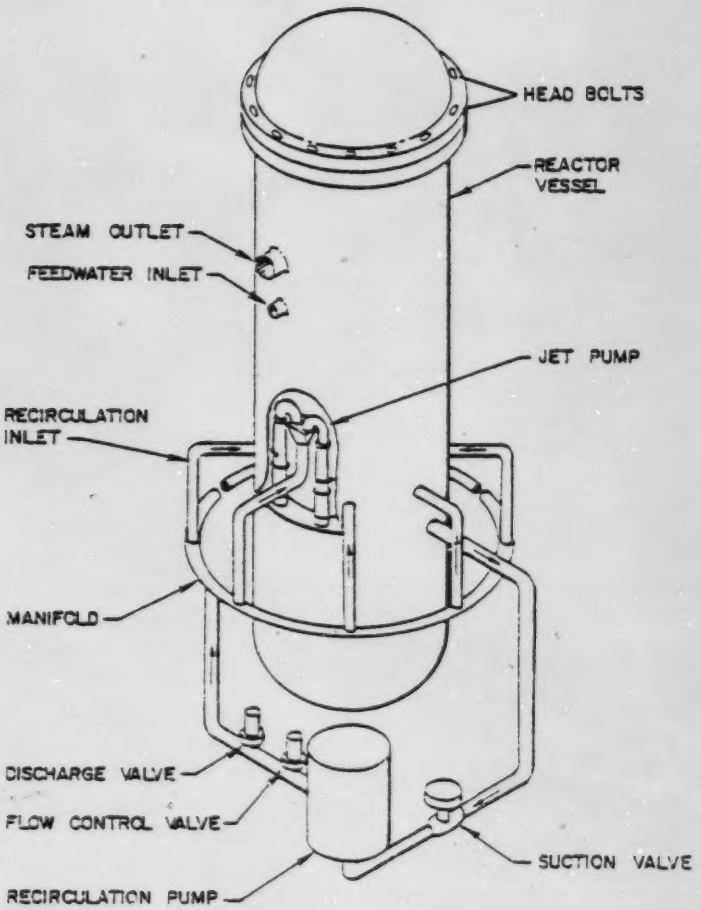

Figure 7-6-4.
BWR vessel arrangement for jet pump.

This flow is discharged into jet pump nozzles, providing driving force for operation of jet pumps. In a typical BWR, there are two identical recirculation loops, each consisting of a constant-speed recirculation pump, a flow-control valve, two isolation valves, a manifold, five inlet risers and ten jet pumps.

JET PUMPS

Each jet pump consists of five basic parts (Figure 7-6-5):

Inlet nozzle.
Driving nozzle.
Suction inlet.
Mixing assembly.
Diffuser.

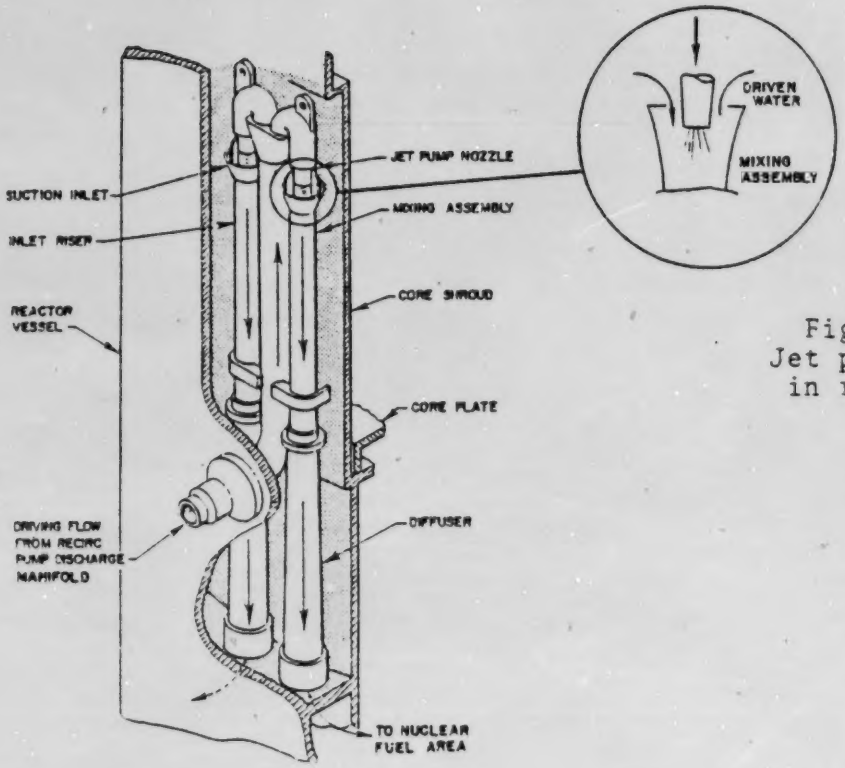

Figure 7-6-5.
Jet pump immersed in reactor core area.

Recirculation pump discharge flows up the inlet riser and down through the jet pump nozzles. As this driving flow is constricted by the nozzle, its velocity increases and pressure decreases. This decrease creates a low-pressure region at the jet pump suctions, which draws surrounding water into the jet pumps. This water (called driven flow) makes up about two-thirds of the total jet pump and core flow. Both driving and driven flows come together in the mixing assembly to reduce possible temperature variations. Then they pass, as a single flow, to the diffuser. There, fluid velocity decreases, resulting in a pressure increase, which forces water into the lower core region and up into the core (Figure 7-6-6).

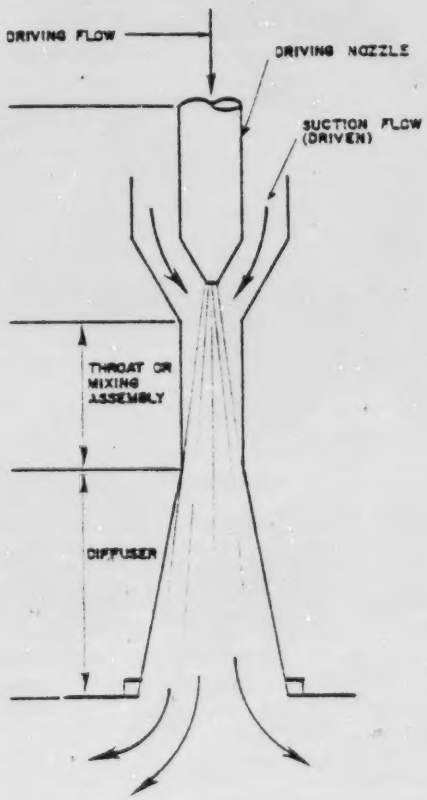

Figure 7-6-6. Jet pump principle.

CONTROL-ROD DRIVE

The control-rod drive system positions reactor control rods in the core to provide power generation, distribution and control, and rapid reactor shutdown (scram) when required. These rods, which contain neutron absorber, enter the reactor vessel through the bottom. They are vertically mounted and hydraulically positioned.

Bottom-entry rods are used for two reasons. First, steam bubbles formed in the top of the nuclear core are poor moderators and allow many neutrons to escape from that area without causing a fission event. Thus, if control rods were top entry, they would remove even more neutrons in that region, causing the axial neutron flux profile to shift further toward the bottom of the core (Figure 7-6-7). But, by using bottom entry rods, axial flux profile is much flatter and average power levels are higher.

Second, bottom-entry rods ease refueling through the reactor top, because only the vessel cover, steam dryers and moisture separators have to be removed.

In a typical BWR, there are up to 185 separate control rods, each with its own drive mechanism (Figure 7-6-8). Hydraulic pressure is supplied to a double-acting, driven piston to position drive mechanism and rod at precisely spaced intervals (called notches). Each notch is six inches apart on the 12-foot long drive mechanism. Without hydraulic pressure, rod and drive assembly weight causes it to settle down onto a notch, where a spring-loaded mechanism maintains the drive position.

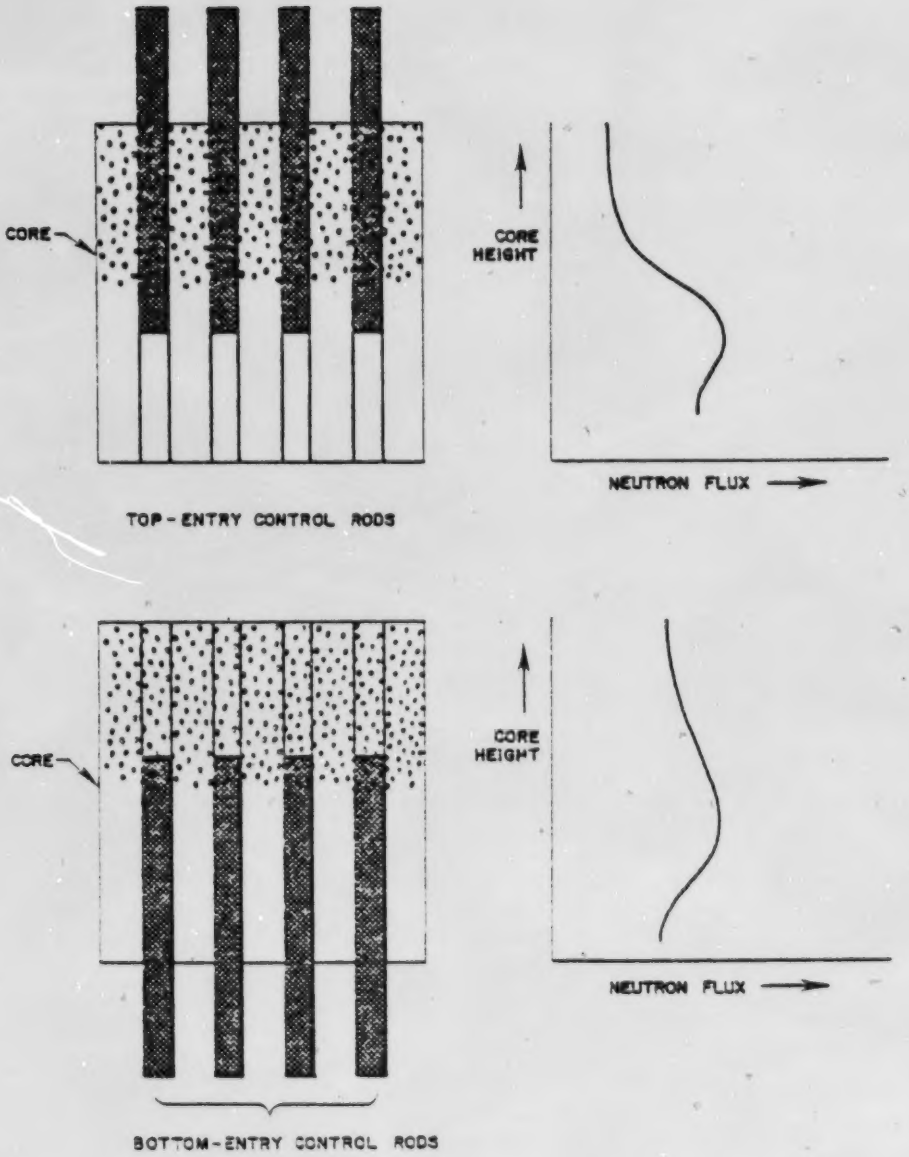

Figure 7-6-7. Neutron flux profiles.

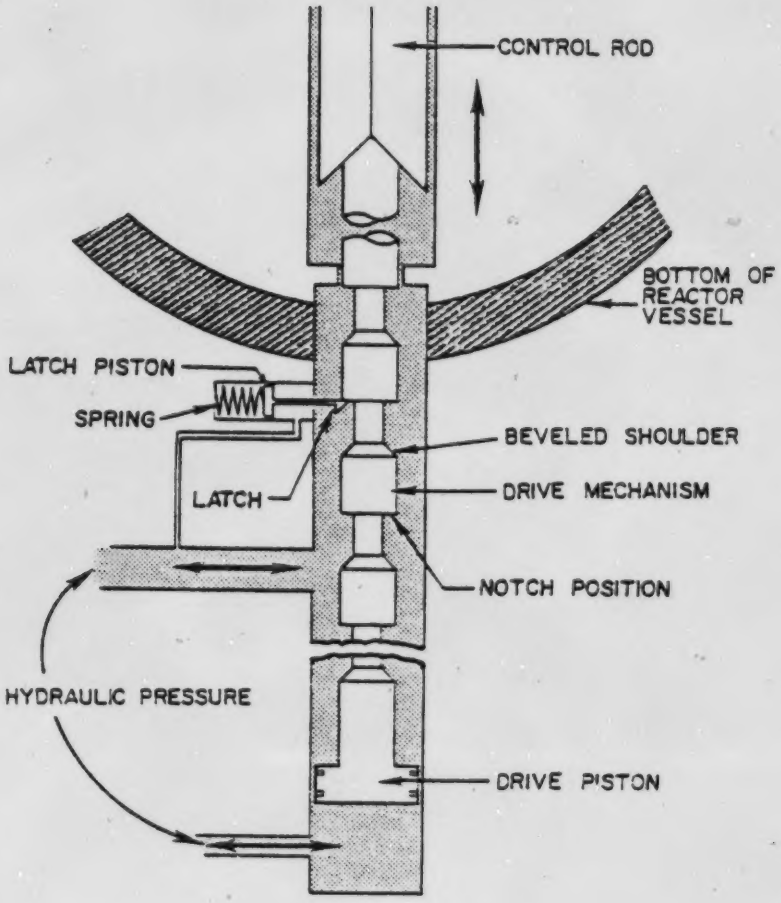

Figure 7-6-8. Control-rod drive mechanism.

To insert a rod, hydraulic pressure is applied to the underside of the drive piston and removed from the top of the piston. The drive mechanism, with rod attached, moves upward into the core. A beveled shoulder on each notch forces the latching device to ratchet so that it does not affect upward rod motion. When hydraulic pressure is removed, the drive settles back onto the latch mechanism. A rod can be moved one or more notches at a time, depending on rod motion desired; however, in most BWR's, only one rod can be moved at a time.

To withdraw a control rod, the latching mechanism must first be ratcheted outward by inserting drive mechanism a short way, which causes lower beveled shoulder of drive to force the latch out. Hydraulic pressure is then applied to the latch piston, keeping it retracted. Simultaneously, hydraulic pressure is applied to the top of the drive piston and removed from the underside. Drive mechanism is withdrawn from reactor until hydraulic pressure is removed. Drive then settles onto latch at its new position.

Rapid insertion (scram) of drive mechanism is accomplished by admitting high-pressure water to underside of drive piston and venting top of piston to atmosphere (Figure 7-6-9).

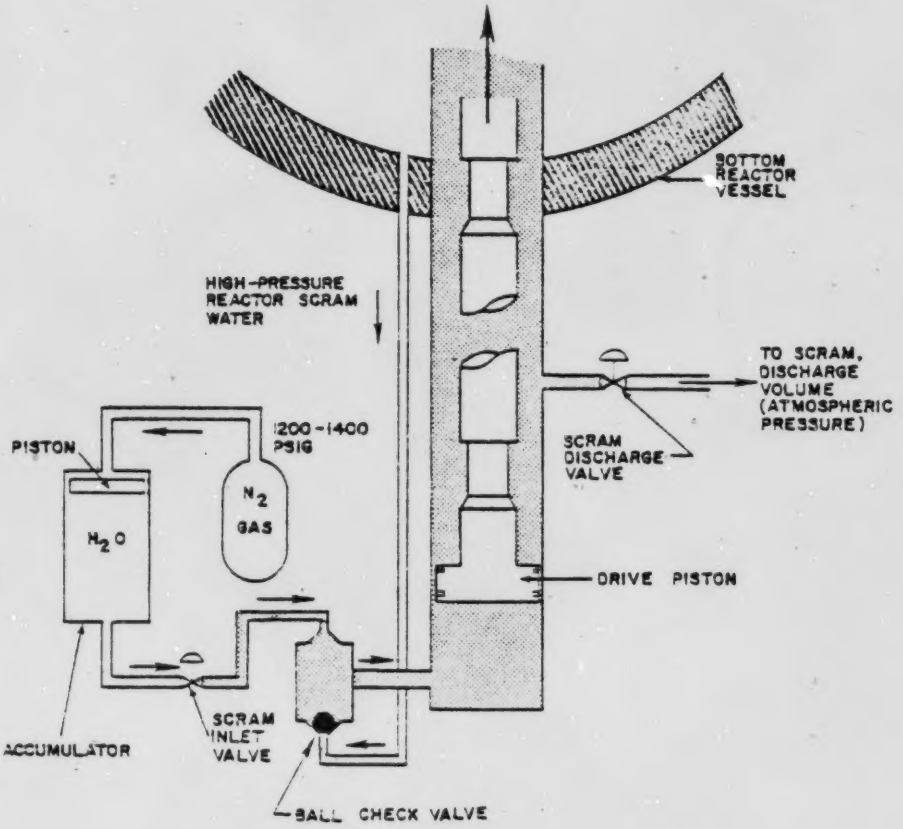

Figure 7-6-9. Control-rod drive (scram) arrangement.

High-pressure water comes from two sources. First, each drive mechanism's hydraulic supply contains a water accumulator charged with nitrogen gas. This accumulator provides stored energy, which is released when scram inlet valve is opened. Simultaneously, scram discharge valve opens, allowing water on top of drive piston to discharge to atmospheric pressure. The large pressure differential across the drive piston rapidly inserts drive mechanism.

The second source of high-pressure scram water is in the reactor vessel itself. A ball check valve provides backup if accumulator pressure falls below reactor pressure. Reactor pressure, on the underside of the ball check valve, forces the ball upward, admitting high-pressure reactor water to bottom of drive piston. Scram discharge valve opens, venting top of drive piston to atmosphere, and drive rapidly inserts into fuel core.

During scram stroke, the latch mechanism ratchets as the drive travels inward. When the drive reaches its full inward travel, the latch mechanism prevents the rod from "rebounding" and locks it in place.

Control-rod blades (Figure 7-6-10) are cruciform (cross) shaped and enclose many thin tubes containing boron carbide (B_4) powder. Most of the boron in the powder is $^{10}_{5}B$ and absorb neutrons according to the following reaction:

$$^{10}_{5}B + ^{1}_{0}n \rightarrow ^{7}_{3}Li + ^{4}_{2}He.$$

Withdrawal of control rods from the core region removes some of the $^{10}_{5}B$, absorbing fewer neutrons and allowing more fission events to occur, and power level increases. Inserting

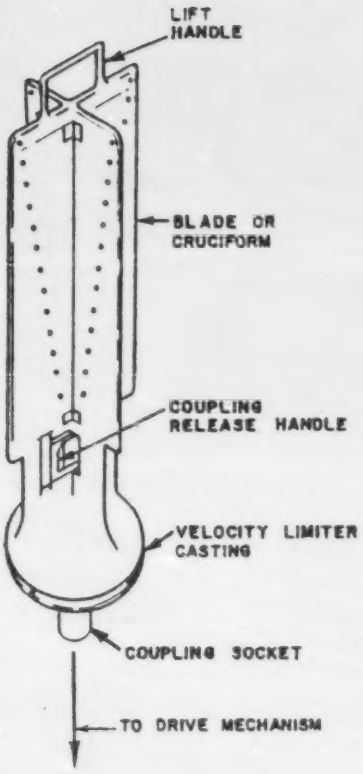

Figure 7-6-10.
Control rod.

control rods into the core reduces power level.

Each control rod is surrounded by four "bundles" of fuel rods (Figure 7-6-11). These rods are about 0.5 inches in diameter and 12 feet long. A typical bundle contains 64 of these rods arranged in an 8 by 8 array and held in place by upper and lower tie plates. Fuel rod spacers every 20 inches provide stiffness and support for the slender rods.

Each rod contains hundreds of small pellets of uranium oxide fuel (UO_2). The pellets are cylindrical, about one-half inch in diameter and in length, and are slightly enriched (2.5%) in uranium-235. The ^{235}U absorbs neutrons to produce fission events, which release more neutrons and generate heat.

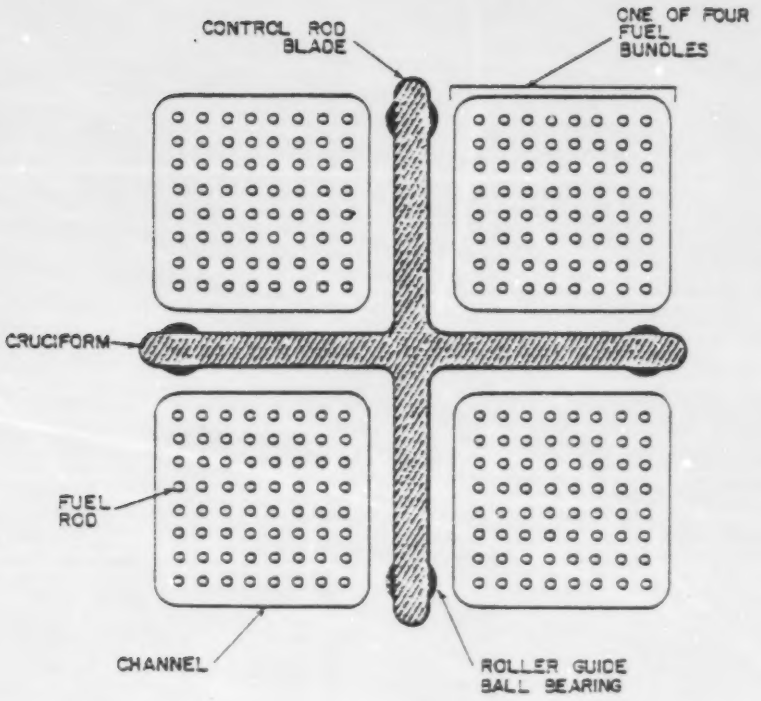

Figure 7-6-11. Fuel bundle/control-rod arrangement.

Reactor coolant, which enters the fuel bundle through the bottom of the lower tie plate, is guided upward along fuel rod surfaces by a fuel channel. Heat, from the fission process occurring in the fuel pellets, is transferred by conduction to the fuel-rod surfaces where, by convection heat transfer, coolant temperature is increased. As coolant travels further into the nuclear core, its temperature continues to increase until it reaches saturation temperature for the pressure at which the reactor is operating.

The primary method of heat transfer then becomes nucleate

boiling, where steam bubbles form in small crevices, or sites of nucleation, on the surface of the fuel rods, detach and are swept away.

During normal power operation, steam bubbles formed by nucleate boiling, unite to form larger bubbles and exit the core as "slugs" of steam vapor (Figure 7-6-12). Steam bubbles are also called voids, because of the relative absence of water molecules, which act as neutron moderators to slow neutrons down so they can be absorbed by ^{235}U to cause more fissions. As will be covered later, these steam bubbles, or voids, are very important in BWR power control.

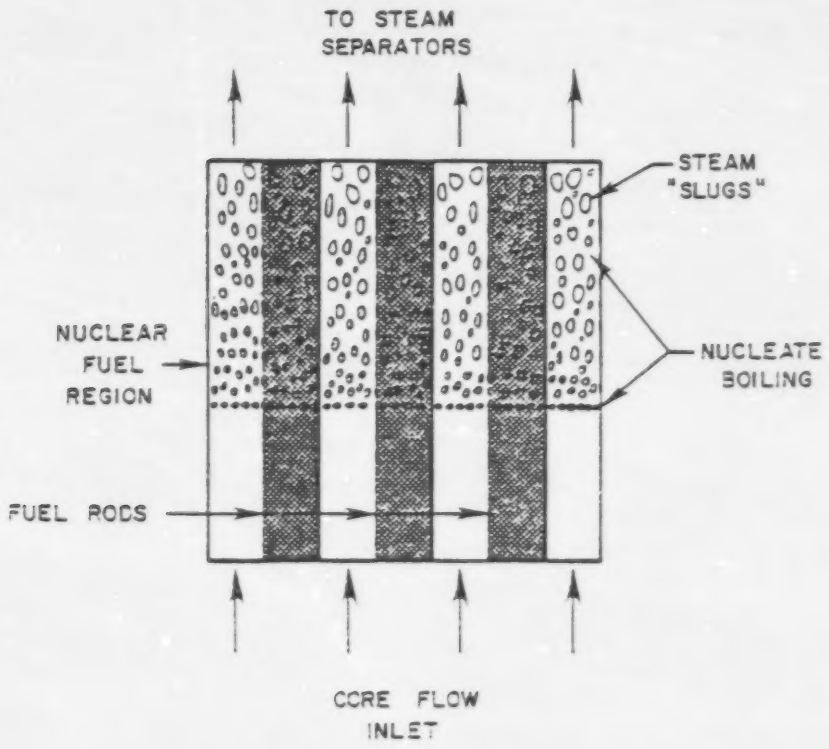

Figure 7-6-12. BWR core characteristics.

STEAM SEPARATION AND DISTRIBUTION

The steam-and-water mixture leaves the core region and passes through the steam-separator assembly. The steam separators (Figure 7-6-13) give a whirling action to the

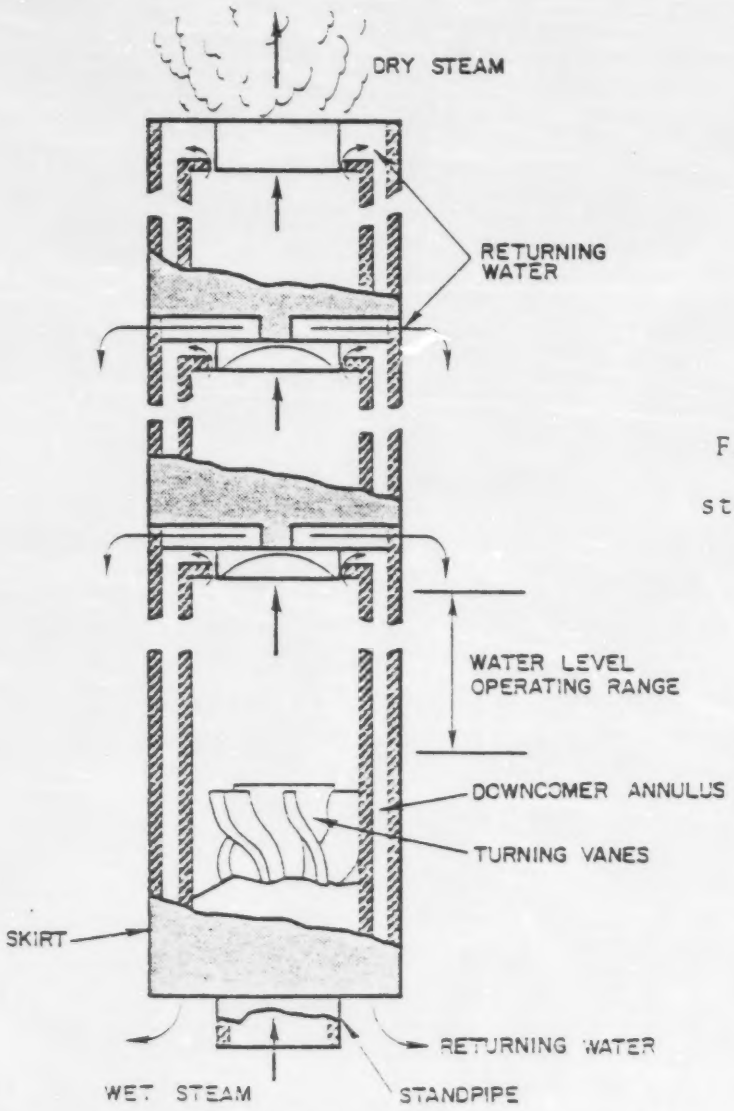

Figure 7-6-13. Internal steam separator.

mixture by means of turning vanes. Much of the water, being heavier than steam, is spun to the outside of the separator and returned to the core. Wet steam is further processed by a steam dryer assembly (Figure 7-6-14), which forces steam to change directions many times. Water is removed by moisture collection "hooks" and drained back to the downcomer region.

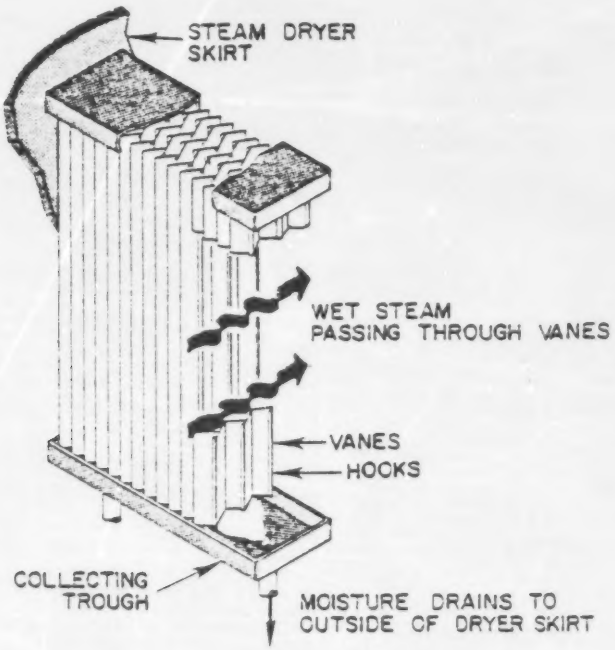

Figure 7-6-14.
Steam dryer.

The main steam system (Figure 7-6-15) then transports the dry steam from the reactor vessel, through the containment, to the high pressure turbine. The system also supplies steam to various other plant components, such as steam jet air-ejectors, turbine-driven reactor feed-water pumps and the turbine gland seal system.

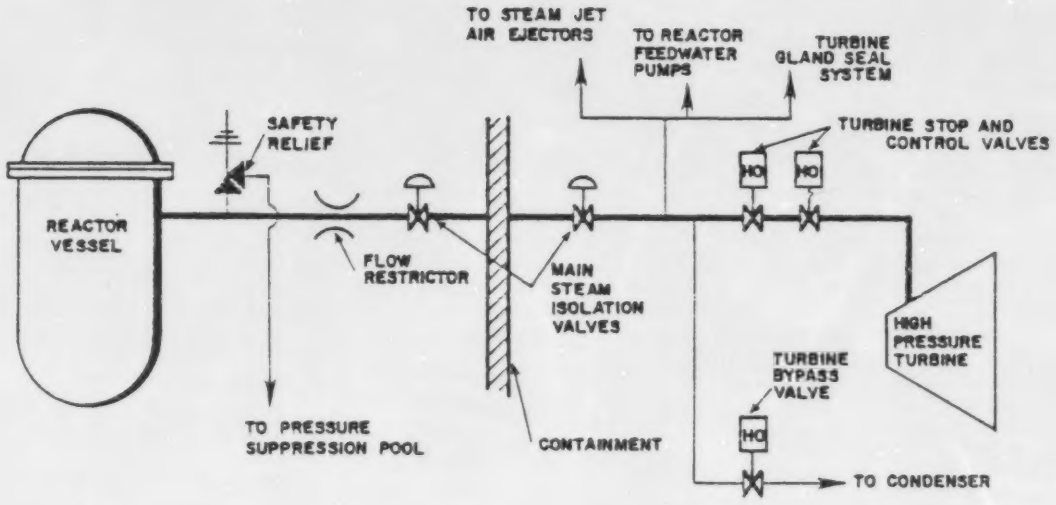

Figure 7-6-15. Main steam system.

If the turbine is not operating or cannot accept all of the steam supplied by the reactor, turbine bypass valves open and send steam directly to the main condenser.

BALANCE OF PLANT

Turbine generators, used for BWR's are similar to those used by PWR's and fossil-fueled plants. One generally consists of a high-pressure stage, a moisture removal device and several low-pressure stages.

REACTOR POWER

Reactor power is basically controlled in two ways:

By moving control rods.

By varying recirculation flow rate.

Power control by control-rod movement is fairly straightforward. Withdrawal of control rods allows more neutrons to cause fission events, which increases reactor power level. Conversely, insertion of control rods removes neutrons from the fission cycle and power level decreases.

Power control by recirculation flow change is accomplished in the following manner (Figure 7-6-10).

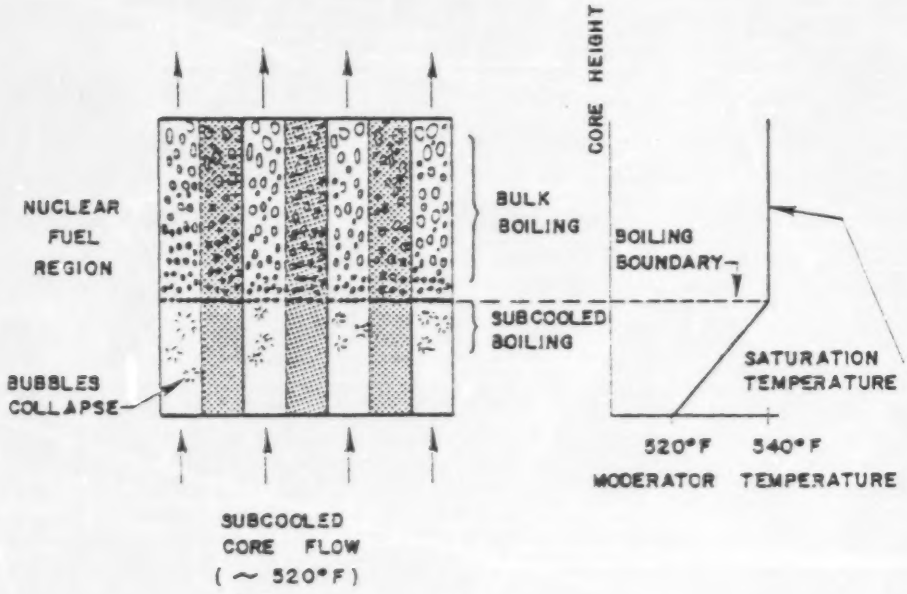

Figure 7-6-16. Conceptual model of recirculation flow control (steady state operation).

Water entering the core is normally subcooled about 20°F by feedwater. Bubbles formed by nucleate boiling in

the lower region of the core collapse when detached from fuel rods by subcooling (subcooled boiling).

As the water progresses up through the core, it is heated until it reaches saturation temperature for the operating pressure.

Bubbles formed from nucleate boiling, which now have sufficient energy to sustain themselves, unite to produce larger bubbles (bulk boiling). Separation between bulk boiling and subcooled boiling is called "boiling boundary."

LOSS OF COOLANT ACCIDENTS

There is a very small probability that certain failures could occur in piping connected to the nuclear steam supply system. There are many pipes, but the largest and most severe break could occur in the suction line of the recirculation system. A guillotine rupture of this pipe is called a design basis accident (DBA) (Figure 7-6-17).

In the unlikely event of a DBA, radioactive water and steam begin flowing out of the reactor, lowering the water level in the reactor vessel and, eventually, uncovering the fuel region. The reactor scrams but heat is still being generated by radioactive decay of fission products in the fuel pellets. If no action is taken, and the fuel is allowed to remain uncovered, this decay heat causes excessive fuel temperatures and possible meltdown of the core.

To limit the probability of such an event, exhaustive measures are taken to ensure the integrity of piping connected

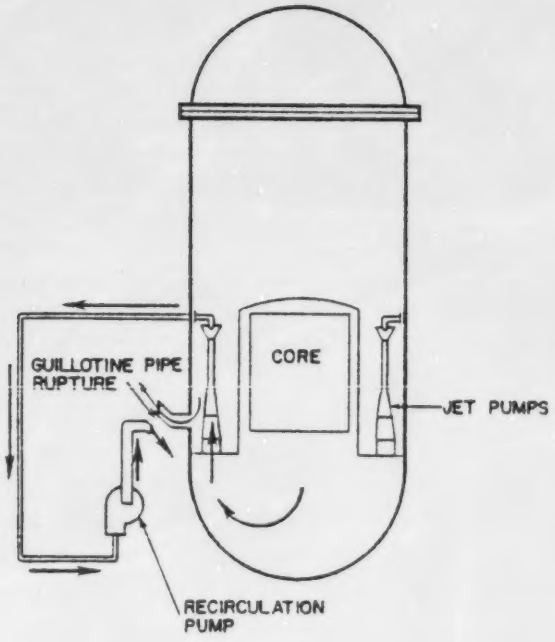

Figure 7-6-17.
BWR design
basis accident.

to the reactor vessel. Piping and welds are inspected periodically by quality-control personnel, using radiography, ultrasonic and dye-penetrant testing methods. Extensive instrumentation monitors the reactor area for changes in temperature, pressure, radiation, humidity, gases and water leakage. These sensitive instruments are designed to detect very small cracks or leaks in piping. Finding and repairing cracks while they are small prevents larger failures from occuring.

If, despite preventive measures, failure occurs, the emergency core cooling system (ECCS) prevents the nuclear fuel from reaching excessive temperatures. The ECCS, which provides this protection by at least two completely independent and automatically actuated cooling systems, is designed to function even with a complete loss of all off-site power supplies.

The emergency core cooling system is really four systems:

High-pressure core spray system.

Low-pressure core spray system.

Low-pressure coolant injection system.

Automatic pressure reducing system.

The ECCS prevents excessive fuel temperatures from occurring following a DBA; however, the radioactive steam and liquid escaping from the break must be contained and prevented from getting into the environment.

CONTAINMENT CONCEPT

The general BWR containment concept is shown in Figure 7-6-18. Multiple barriers, designed to prevent or limit the release of radioactive fission products following an LOCA include:

Fuel pellet matrix (ceramic UO_2) which resists decomposition, erosion and leaching, even when exposed to a water environment.

Fuel pellet (and the majority of the fission products) cladding (sealed metallic tubes).

Steel reactor pressure vessel, with 6-inch thick walls, surrounding the fuel.

Primary containment (surrounds reactor vessel and limits release of fission fragments into secondary containment).

The primary containment structure, a free-standing, steel, pressure vessel designed to meet ASME pressure vessel

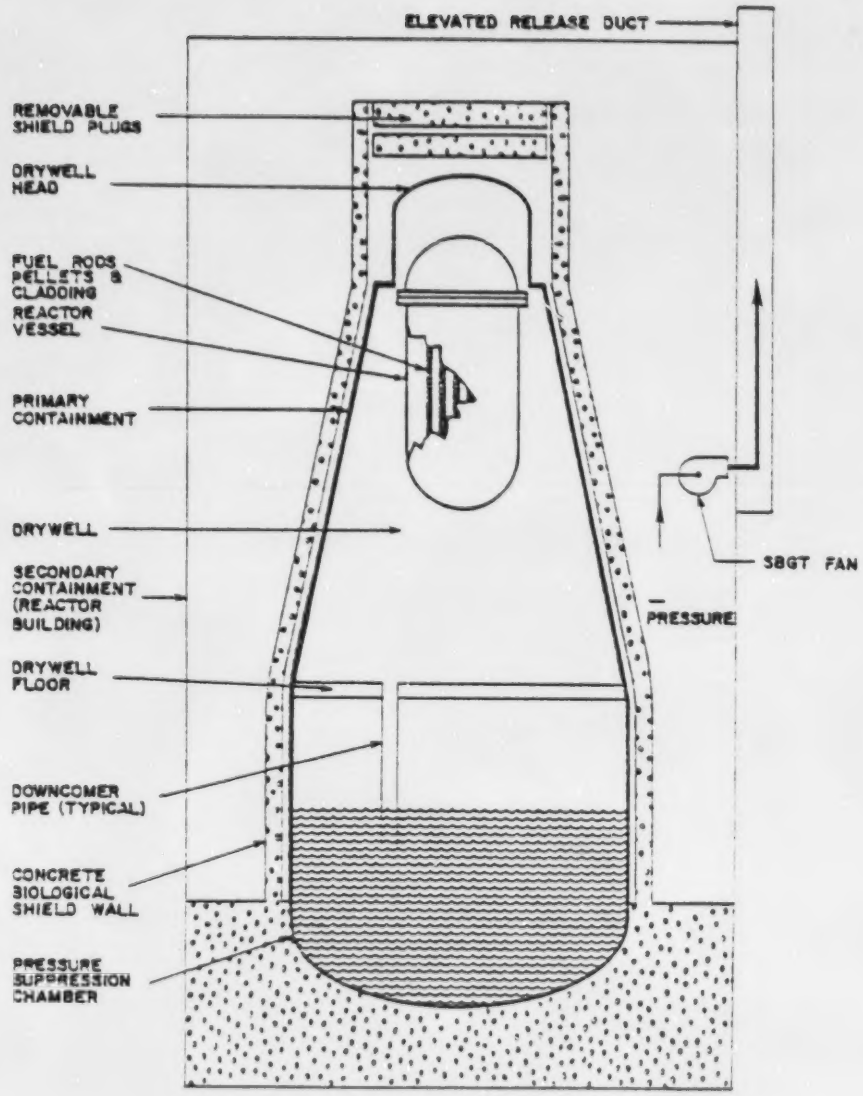

Figure 7-6-18. BWR containment concept.

codes, is divided into two areas called the "drywell" and the "pressure-suppression chamber."

The drywell, in the shape of a truncated cone with a cap, houses the reactor and its associated primary systems. The main purpose of the drywell is to contain the effects of a DBA and direct released steam into the suppression chamber pool via downcomer pipes. The drywell floor serves as a pressure barrier between the drywell and the suppression chamber.

The primary function of the pressure suppression pool is to provide a reservoir of water capable of condensing steam flow (from the drywell) and reducing the pressure transient associated with a DBA. Also, it serves as a principal source of water for various emergency core cooling systems.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

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1. Describe, with appropriate sketches, the purpose and operation of the following items in the nuclear steam supply system (NSSS):

   Nuclear boiler and internals.

   Recirculation system.

   Control-rod drive system.

Main steam system.

2. Describe the processes involved in BWR power control, including a description of the effect of control-rod movement and recirculation flow changes:

NUCLEAR TECHNOLOGY

MODULE 7-7 REFUELING OF REACTORS. -

GERALD C. WALLACE

REACTOR
OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER – SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 7. REACTOR OPERATIONS

MODULE 7-1 Reactor Operator Licenses

MODULE 7-2 Administrative Control for Nuclear Systems

MODULE 7-3 Reactor Operating Experiences

MODULE 7-4 Reactor Startup Procedures

MODULE 7-5 Pressurized Water Reactor Operating Characteristics

MODULE 7-6 Boiling Water Reactor Operating Characteristics

MODULE 7-7 Refueling of Reactors

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Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

The refueling of large, modern power reactors has proven to be the most dynamic activity associated with normal reactor operation. Current practice is to shut down once a year for a period of approximately two months to refuel the reactor, perform required maintenance and do periodic surveillance testing. The refueling process requires a large number of people, a great deal of coordination and communication and, if not done quickly and professionally, results in excessive delay and cost.

Handling of new and used fuel is relatively simple and quite safe as long as personnel understand this equipment and the interlocks and procedures.

MODULE PREREQUISITES

The student should have completed all previous modules in this Course.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List all major steps taken after reactor shutdown, prior to actual refueling.
2. Describe function and operation of important interlocks associated with each of the following pieces of refueling equipment:

   Bridge and trolleys.
   Fuel grapple and hoist.
   Control-rod grapple and hoist.
   Fuel-transfer system and upenders.
   New-fuel elevator.
   Neutron instrumentation.

3. List principal steps taken during PWR refueling.
4. List primary differences between BWR refueling and PWR refueling.
5. List major refueling activities taken after fuel shuffle but prior to reactor startup.
6. Describe three other related activities accomplished during refueling outage.
7. List major operations performed before spent fuel can be shipped off site.

NEW FUEL RECEIPT

New-fuel assemblies, to be inserted into the core, arrive at a plant about three months prior to scheduled shutdown. Each one must be inspected for defects, shipping damage and proper documentation before being inserted into new-fuel storage racks. They arrive in casks, each having two fuel assemblies (and documentation as to the type fuels, enrichment, serial number, codes and standards under which each was manufactured).

The casks are oblong steel boxes, about 15 feet long, 3 feet wide and 4 feet high. Each cask has inspection windows through which it is possible to see accelerometers to determine if the fuel has been subjected to excessive stress loading in any direction. If the cask is bumped or dropped, small flags trip, if a flag is showing, the fuel should be returned to the vendor without opening the cask.

The top of the cask is simply unbolted and lifted off with a crane to expose the fuel. The fuel lacks sufficient rigidity and strength to be handled in other than the vertical position without full-length support. This support is supplied in the cask by a "strongback" arrangement (Figure 7-7-1).

Fuel is clamped to a strongback, which pivots at the bottom so that the weight of the fuel is always supported as it is raised from the horizontal to the vertical position.

The cask is provided with detachable "outrigger" arms, which give additional support to ensure that the cask does not topple while fuel is raised and removed. Once fuel is

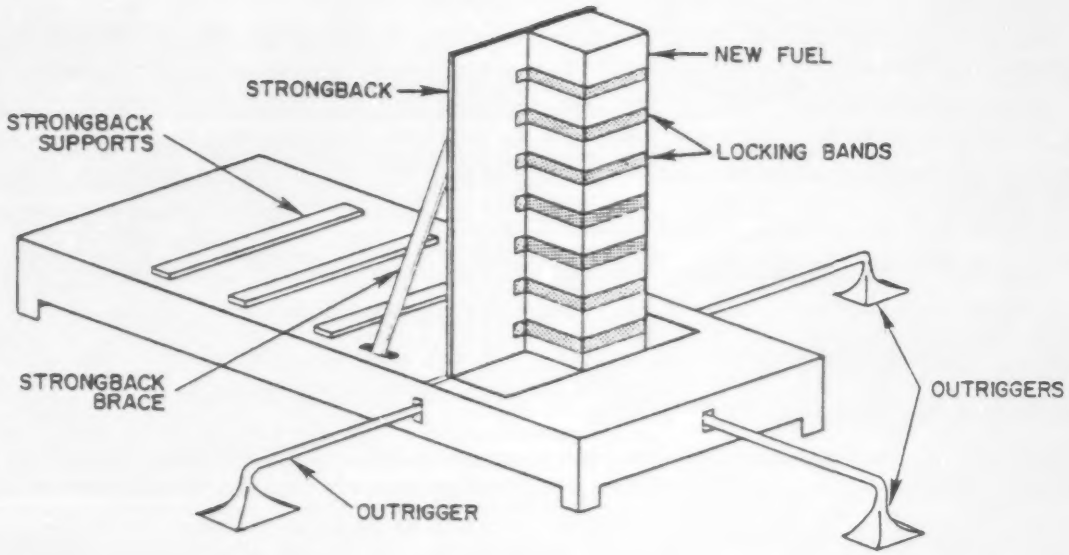


Figure 7-7-1. New fuel cask.

vertical, the strongback is locked into that position, and the crane used to rotate the assemblies is made ready to actually lift the fuel from the cask.

The overhead crane is rigged with a grapple to lift the fuel assembly and a cell to indicate weight of load being lifted. The grapple has four pivoting fingers that engage in the upper-end fitting of the fuel assembly so it can be lifted free of the cask.

On the A side (Figure 7-7-2), the lifting cylinder is raised by rotating the threaded cylinder screw. As the lifting cylinder is raised, pivot arms are rotated, causing

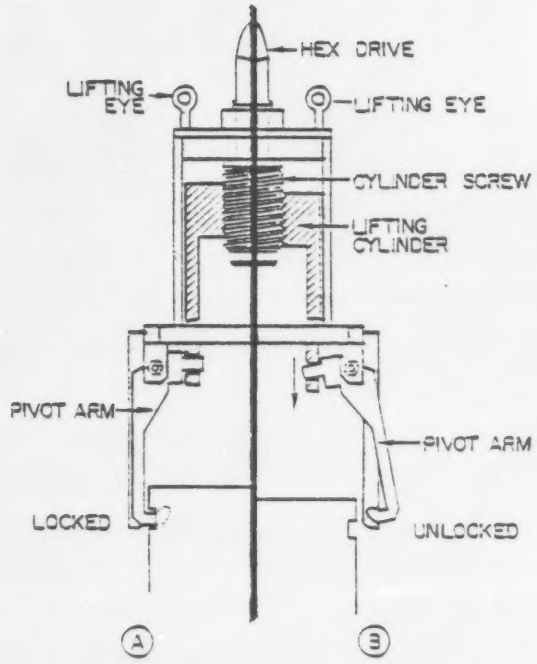


Figure 7-7-2.
New-fuel handling tool.

grasping fingers to move inward, locking into slots at the top of the fuel assembly. When lifting cylinder is lowered, grasping fingers move outward, releasing the fuel assembly.

During vertical fuel movement, the load cell must be observed for high or low weight, which indicates assembly is binding on something and must be freed before proceeding. This is true, whether on the crane or on a fuel-handling bridge: overload stops up motion; low load stops down motion.

Once the assembly is free of the cask, it can be taken to an area designated for fuel inspection. At this point, the assembly is checked visually for defects and serial number verification before being placed in storage. It may be placed in dry storage or in the new-fuel elevator and lowered into wet storage for transfer to a storage rack via the fuel-handling bridge.

Approximately one-third of the core is removed each year, so the above operations must be performed for each new assembly entered. Accurate records of each assembly must always be available. Personnel in charge must be able, at all times, to locate and identify each assembly by number.

REACTOR SYSTEMS PREPARATIONS

The first phase of preparation is draining water from the reactor coolant system to a level low enough to allow removal of the head. While this is going on, preparation for actual head removal is being made. Once the loop is sufficiently cooled, insulation can be removed and stored in its assigned location.

The next step is detensioning of studs that secure head to reactor vessel. These studs, stretched approximately 50 mils to adequately seat the head, must be relaxed to unscrew them from the vessel flange. A tensioner (Figure 7-7-3) sits on the head, and the stud fits inside the tensioner. The upper threaded portion of the stud is threaded into a lifting cylinder housed in the tensioner. Hydraulic pressure is then applied to the piston, and the stud is physically stretched several mils. The studs are about 6 inches in diameter, so it takes several thousand pounds of force to stretch them.

The stud removal tool (Figure 7-7-4), is a long threaded shaft, which is pinned to the stud and turned by air to unscrew the stud.

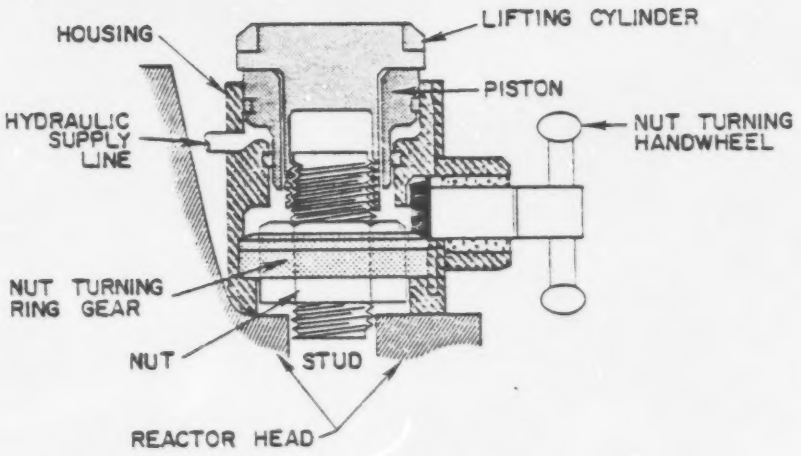
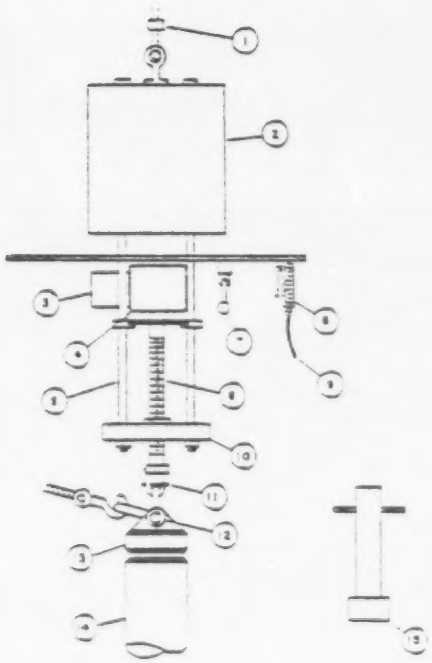


Figure 7-7-3. Typical stud tensioner.



1	Stud Hoist Hook
2	Balance Spring Support and Shield Assembly
3	Motor
4	Motor Reducer
5	Stud Handling Tool
6	On-Off Valve
7	Direction Switch
8	Drive Screw
9	50 cfm, 30 psi Air
10	Support Bracket
11	Stud Hook
12	Transfer Yoke
13	Stud Handling Adapter
14	Stud
15	Stud Chaser and Seal
	Plug Tool

Figure 7-7-4. Typical stud removal tool.

Another operation concerning the head is the uncoupling of service lines, including electrical connections to the control rod mechanisms, instrument leads to the position indicators for the rods, cooling water to the control rod mechanisms and possibly power leads to the cooling fans for the head. This work is normally done while the studs are being detensioned.

Once this has all been accomplished, the head is ready to be removed. The rigging is matched up with the head and the crane centered over the head. The head is removed and taken to its storage stand, where it will remain until reinstallation on the vessel.

The last major equipment operation involves removal of the reactor vessel upper internals, which have provided alignment and support for the core. (This operation requires a different crane rig and precision lift components.) These internals are slowly lifted and stored under water in the refueling canal.

Once the internals are stored, the pool can be flooded to the proper level with borated water, with final equipment and personnel checks made before actual fuel shuffle begins.

FUEL-HANDLING EQUIPMENT

Most present day refueling devices are highly interlocked and remote controlled. Their operation should be thoroughly understood before attempting to handle fuel.

BRIDGE AND TROLLEY OPERATION

The main structural part of fuel-handling equipment is the fuel-handling bridge (Figure 7-7-5). It consists of a

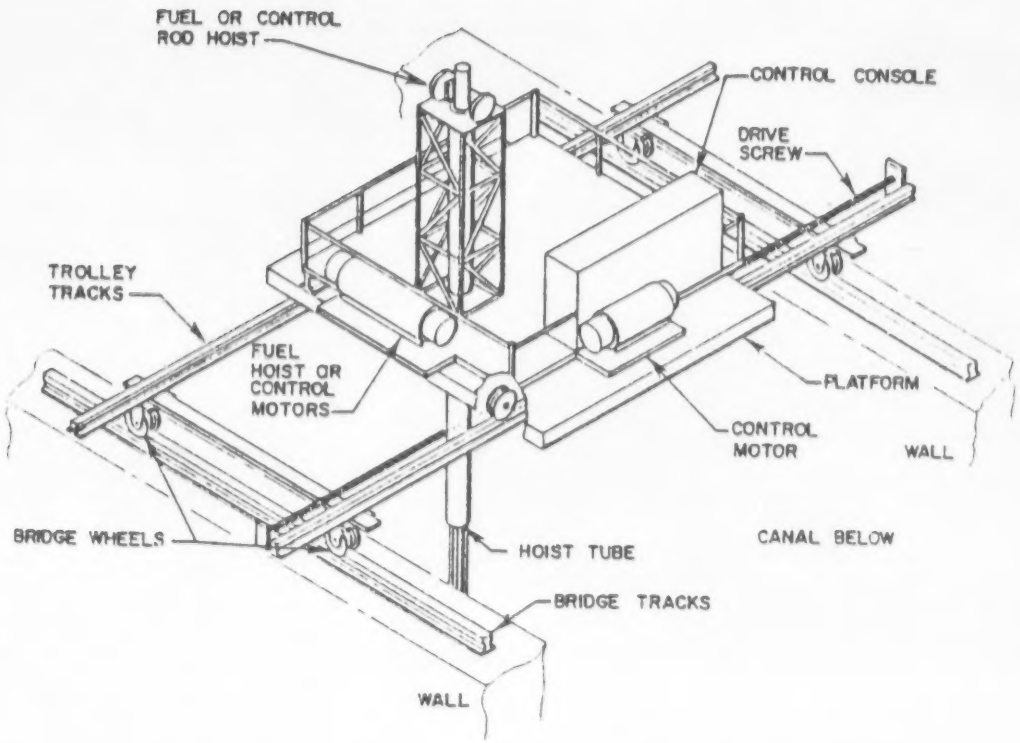


Figure 7-7-5. Typical fuel-handling bridge.

rectangular framework of steel beams which have wheels and roll on rails over the fuel-storage pool or reactor vessel (a bridge is located at each of these locations, and some

installations have an auxiliary bridge in the reactor building to help with the shuffle).

Each bridge has rails running its full length to support the trolley, which travels in a perpendicular relation to the bridge travel. Fuel and control-rod hoists are attached rigidly to the trolley and can be moved to any location by manipulation of controls associated with the motor-driven bridge and trolley.

Bridge position must be precise to ensure that fuel is inserted or withdrawn with minimal contact with surrounding assemblies or storage-rack sides. Generally, the bridge is positioned first by rotating a pistol grip handle in the direction of desired travel. This control, which provides smooth acceleration as the handle is rotated, should be rotated smoothly up to full speed and smoothly decelerated and stopped at desired position.

Time is critical during refueling, and bridges should be operated at full speed, except at the beginning and end of travel when sudden starts and stops, which could jar the fuel are avoided.

Interlocks are provided for both the bridge and trolley to ensure that fuel is not damaged while being handled. The first interlock prevents operation of any control while another is in use. This interlock ensures that motion is allowed in only one direction at a time to minimize the possibility of fuel damage. Also, each part has overtravel limits in each direction, whicl stops before the mast hanging down from the trolley comes in contact with a pool wall, alignment stud or any other obstruction that could damage fuel or equipment.

Some things are not protected by interlocks. The operator must be aware of such possibilities before any motion is attempted.

The most hazardous object is even less obvious - other personnel. When refueling is being done, many people are in the area and could easily be injured by the heavy equipment. Strict controls must be set up for personnel safety, and the operator must be alert for people in the operating zone of the bridge.

FUEL HOIST AND GRAPPLE

The fuel assemblies are lifted by an electric hoist (with a mechanical grapple on the cable end) and pulled into a stainless steel mast, which provides alignment and protection (Figure 7-7-6). The hoist motor has a control (identical to bridge and trolley controls) and also has provisions

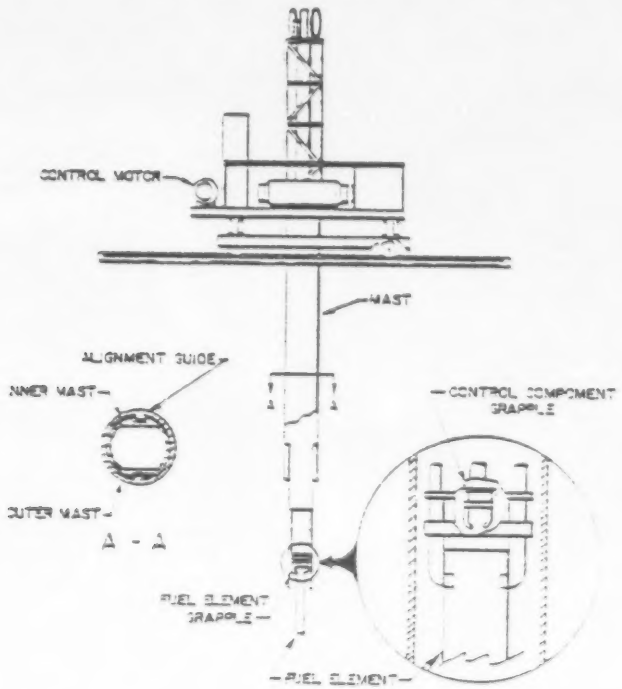


Figure 7-7-6. Main fuel-handling bridge with multiple-function mast.

for manual raising or lowering. The control handle is located on the control console. The vertical position of the grapple is noted by readout of a metal tape, which is attached to the grapple, and by limit-switch-operated lights at normal stop positions of the grapple (such as "fully up" or "fully down").

A most important instrument used is the load cell, which measures the weight of the grapple and fuel. When fuel is being raised or lowered, the operator's eyes should be on the load cell, and the motion should be stopped if the weight changes, indicating a possible stuck assembly.

Interlocks are provided for overload and underload to protect the fuel if it should stick in either direction, but an alert operator always stops the hoist before the interlock has a chance to do so.

The grapple is a hydraulically operated linkage of four fingers that engage slots in the top of the fuel assembly. A serious potential accident is dropping of a fuel assembly; therefore, the grapple is interlocked mechanically and electrically and the fingers cannot be disengaged unless the fuel is supported from beneath (and some of the weight of the grapple itself is supported).

The steel tape is provided to tell exactly how far the hoist has been lowered and should always be observed by the operator when preparing to grapple or ungrapple fuel, or before moving the bridge or trolley after a fuel assembly has been handled. Each time the fuel is to be ungrappled, a visual check should be make of 1) the load cell, 2) an indicating light, which tells that sufficient load has been lost to satisfy ungrapple interlocks, and 3) the elevation tape.

One very important interlock that is easily overlooked is the "hoist up" interlock, which stops the hoist when it

is fully withdrawn. This seems an obvious action to have, but the operator should be aware that, if this limit is exceeded, the water shielding is reduced and radiation levels at the controls increase substantially if the fuel has been used.

CONTROL-ROD HOIST AND GRAPPLE

Operation of the control-rod hoist is essentially the same as that of the fuel hoist; the interlocks are the same except for the weight loss, which is not required or practical. When the hoist is lowered, it simply lowers a tube which mates with the top of a fuel assembly or rod container to provide alignment for a telescopic cyclinder, which actually grapples the rod.

The rod has 16 pins, each 12 feet long and suspended from a spider arrangement. If this tube were not used to align the fuel, it would be almost impossible to have all 16 pins hit their openings.

FUEL-TRANSFER SYSTEM AND UPENDERS

Once fuel is removed from a reactor, it must be sent to storage for radioactive decay, prior to off-site transfer. This transfer, which is complicated considerably by the extremely high radioactivity of the fuel, must be made under many feet of water.

The fuel is lowered into a basket (perforated for cooling) that can be hydraulically rotated from the vertical to a horizontal position (Figure 7-7-7). This basket is attached to a long carriage (of stainless steel tubing) with wheels, which allow the carriage to proceed by rail from the

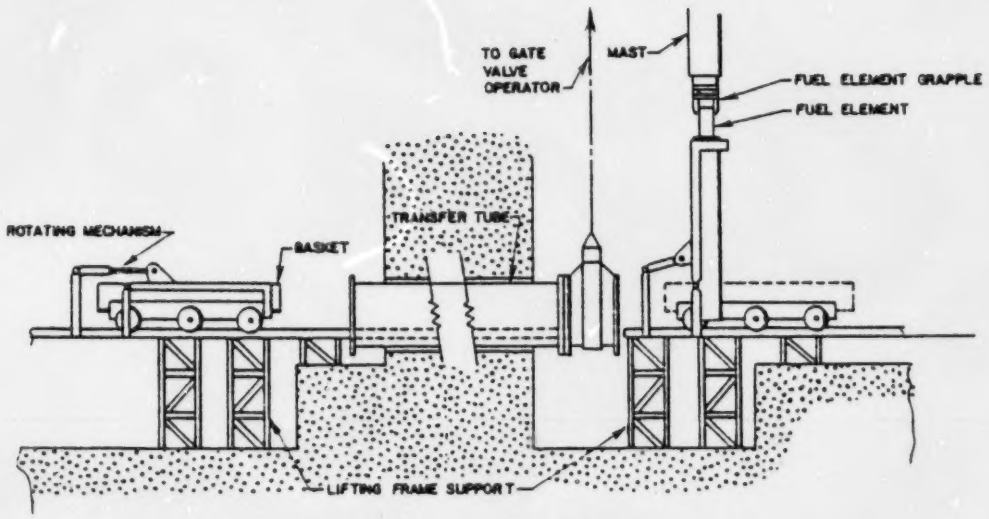


Figure 7-7-7. Fuel-transfer system.

reactor building to the spent fuel pool.

The underside of the carriage has a chain, which engages a drive sprocket (driven by an air motor) to propel the carriage from one building to the other. Operators, stationed at control consoles in each building, control the carriage. If either operator is dissatisfied with the operation, the carriage may be stopped at any time. Each operator is able to rotate the upenders to the vertical or horizontal position. Slow zones, just prior to stopping, in either building ensure an accurate position when the carriage stops.

An interlock arrangement exists between the bridge and carriage. When the bridge is over the upender, the fuel hoist cannot be raised or lowered unless the upender is in the vertical position. Likewise, the upender cannot be rotated up or down while the bridge is over it. These interlocks prevent operation of either device while fuel is partly

enclosed in the basket, which could damage the fuel.

NEW-FUEL ELEVATOR

A very simple device (Figure 7-7-8) is used to get new fuel from the operating floor to the bottom of the pool where it is picked up by the fuel-handling bridge and transferred to a storage rack position. It consists of an electric hoist attached to a stainless steel basket, which houses the fuel.

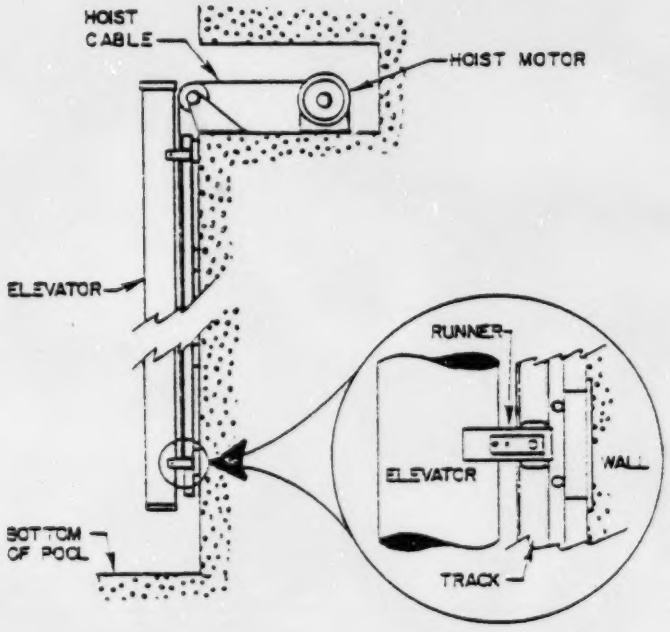


Figure 7-7-8. Fuel-handling elevator.

Pushbutton controls start the elevator or stop it at any elevation. It should be noted that the fuel mast must come directly over the elevator; thus, if the elevator is not fully down, it could be struck by the mast. A mark should be made on the cable to ensure that it is fully down before bringing the bridge over it.

As elevator is raised and lowered, it is guided by rails. (Under no circumstances should spent fuel be placed in the new-fuel elevator. When in a full up position, the spent fuel would be completely out of water, and such a hazardous situation can never be allowed.)

AUXILIARY EQUIPMENT (TELEVISION AND NEUTRON INSTRUMENTATION)

Underwater television has proved useful for fuel handling. Having to keep the fuel under water creates difficulty in detailed examination and identificaition. Irradiated fuel is characteristically black, and the identification numbers are difficult to read without television. Also, any suspected damage can be observed in detail, and valuable videotapes can be made.

Auxiliary neutron detectors may be located in the core to provide more sensitive observations of reactivity effects, especially in an initial fueling. It is not required normally for refueling.

REFUELING CYCLE

Once the head is removed, the internals stored and the water level adjusted correctly, equipment can be checked out and actual fuel shuffle begun. Equipment used for fuel handling must be tested prior to each refueling. These tests consist of interlock checks and actual operation of the equipment, using "dummy" fuel assemblies (replicas of true fuel, dimensionally and by weight). Every crew should use the equipment (with dummy) prior to operation with real fuel.

Fuel in reactors is arranged in zones, according to enrichment; that is, the center one third of the core is of the lowest enrichment, the second zone from the center has slightly higher enrichment and the outer zone has the highest enrichment. In some instances, the two lower enrichment zones may be arranged in a checkerboard pattern. The idea is to equalize burnup as much as possible. The refueling process, then, calls for 1) removal of the center one-third of the core, which has been expended, 2) replacement of those assemblies with those from zone 2, which are now the lowest enrichment, 3) replacement of those from zone 2 with those from zone 3 and 4) placement of new fuel in the outer zone. Plants attempt to refuel each year; thus, each assembly is cycled through each enrichment zone and out of the core in three years.

When the fuel is shuffled, the control rods also must be relocated.

BWR REFUELING

BWR and PWR refueling are essentially the same, except for differences related directly to basic hardware. The advantage of a BWR is its simplicity. This concept is carried over into refueling. Fuel is kept in the reactor building, which eliminates one bridge and transfer system. The remaining bridge is generally more simple and, because the control rods enter from the bottom and are not normally shuffled, the hoist is much less elaborate.

The three-zone, three-year, inward shuffle is used, but many more fuel "bundles" are involved (about 750 versus 200 for comparable, large plants). A grapple picks up the bundles, but only one finger is used; it passes through a diagonal "bail," which is a very simple handle shaped as if it were to be picked up by hand. An indicator tells the operator when the bail is homed into the grapple, and the grapple finger is in position for the bundle to be picked up with certainty.

(Note: Small test reactors are much different and vary too much to elaborate to any extent. Generally, equipment is more likely to be primarily manual and much less sophisticated.)

POST REFUELING ACTIVITIES

Once a new core is intact and verification is completed, internals can be reinserted and the pool pumped down. A significant cleanup operation, involving decontamination of walls and floors of the pool, must be done promptly (even as

the level decreases) to minimize airborne contamination and to ensure maximum cleanliness during reinstallation of the reactor head.

Once the pool is drained, the head can be replaced and studs threaded into the vessel flange, and tensioned. The rods must then be recoupled to the drive mechanisms.

Once service systems (electrical, instruments, cooling water and ventilation) are connected to the head, the reactor is ready for operation. (Note: The plant must be sufficiently pressurized to vent the control-rod drive mechanisms.)

Related activities have been accomplished during the shutdown so that a speedy return to power is possible. These related activities include:

Inservice inspection of reactor coolant pressure boundary.

Reactor coolant and containment isolation valve leak test.

Inspection of reactor coolant pumps, especially mechanical seal.

Inspection of steam generator tubes.

BALANCE OF PLANT ACTIVITIES

A great deal of activity is always present in the secondary plant during refueling. For example, turbine inspection and maintenance, which may be of longer duration than the refueling itself. Also, overhauls of major equipment (that cannot be taken off line at power) are performed

throughout the plant. Packing replacements, oil changes, repair steam leaks, piping reruns, modifications to existing equipment and installation of new equipment and systems all are accomplished to some degree during each refueling.

SPENT-FUEL ACTIVITY

Spent fuel, which remains under water, must be allowed to decay for some time before it can be shipped. It is "sipped" to determine if it has undergone cladding failure before it is shipped; this examination may be made with television, periscopes or other special equipment.

Once spent fuel is ready for shipment, casks are brought to the plant and placed in the pool. Each cask is lowered into the pool and designated fuel is inserted into it. A top is placed on the cask, and the entire unit is removed from the pool, surveyed and decontaminated. Proper documentation is attached to the assembly, and the fuel is transported for reprocessing.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

None.

1. List major steps taken after reactor shutdown and prior to actual refueling:

2. Describe function and operation of each of the following pieces of refueling equipment, including interlocks:

   Bridges and trolleys.

   Fuel grapple and hoist.

   Control-rod grapple and hoist.

   Fuel transfer system and upenders.

   New-fuel elevator.

3. List major steps in PWR refueling procedure; briefly explain any differences from a similar procedure for a BWR:

4. List special features of test reactor refueling:

5. List major steps, after fuel shuffle, in preparation for reactor startup:

NUCLEAR TECHNOLOGY

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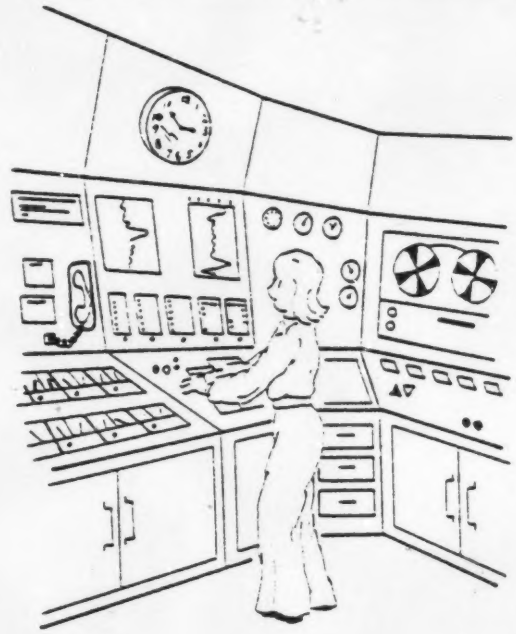

REACTOR
OPERATION

COURSE 7

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER – SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

NUCLEAR TECHNOLOGY SERIES

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for

COURSE 7, REACTOR OPERATIONS (SEVEN MODULES)

NOTE: Each module in this Course is numbered separately, and the page numbers begin with page 1 in each module. Therefore, this Index lists for each entry the module number (underlined), followed by a comma and the page number. Figures and tables are listed by their numbers, followed by the module and page number in parentheses. A Master Index is available in a separate volume. This general index merges all index entries from the 35 Nuclear Courses. It is a valuable research tool, with which the student or researcher can locate all the references to any subject within the 35 Courses.

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